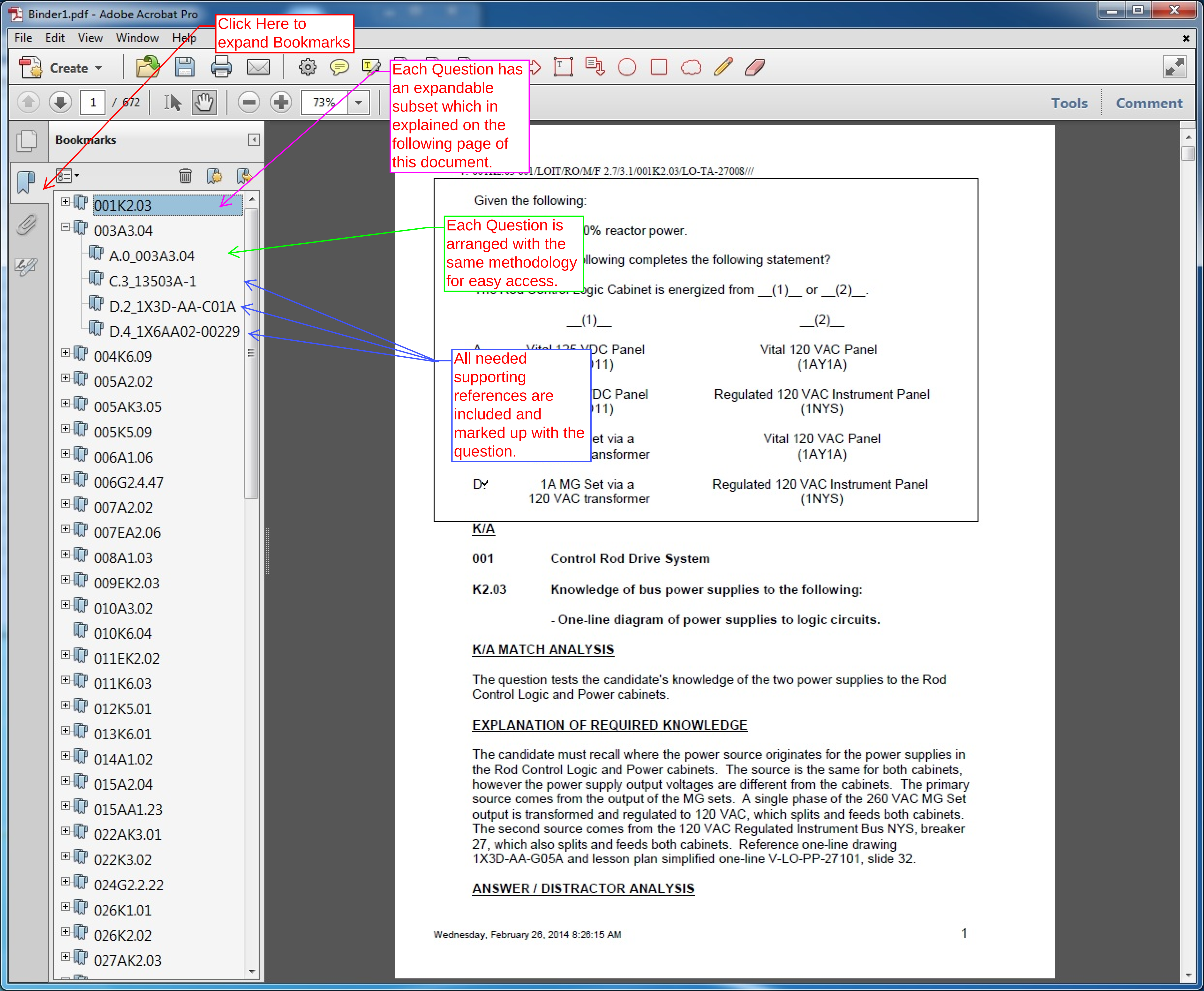


/LOIT/RO/M/F 2.7/3.1/001K2.03/LO-TA-27008///

Given the following:

0% reactor power.

llowing completes the following statement?

gic Cabinet is energized from __(1)__ or __(2)__.

	__(1)__	__(2)__
A.	Vital 125 VDC Panel (…011)	Vital 120 VAC Panel (1AY1A)
	…DC Panel (…11)	Regulated 120 VAC Instrument Panel (1NYS)
	…et via a …ansformer	Vital 120 VAC Panel (1AY1A)
D✓	1A MG Set via a 120 VAC transformer	Regulated 120 VAC Instrument Panel (1NYS)

K/A

001 **Control Rod Drive System**

K2.03 **Knowledge of bus power supplies to the following:**

- One-line diagram of power supplies to logic circuits.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the two power supplies to the Rod Control Logic and Power cabinets.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate must recall where the power source originates for the power supplies in the Rod Control Logic and Power cabinets. The source is the same for both cabinets, however the power supply output voltages are different from the cabinets. The primary source comes from the output of the MG sets. A single phase of the 260 VAC MG Set output is transformed and regulated to 120 VAC, which splits and feeds both cabinets. The second source comes from the 120 VAC Regulated Instrument Bus NYS, breaker 27, which also splits and feeds both cabinets. Reference one-line drawing 1X3D-AA-G05A and lesson plan simplified one-line V-LO-PP-27101, slide 32.

ANSWER / DISTRACTOR ANALYSIS

Wednesday, February 26, 2014 8:26:15 AM 1

Question PDF Bookmark Anatomy

A.0 – Question
- A.1 – Distractor Analysis
- A.2 – Misc Comments/Feedback/History
- A.3 – NUREG ES-401-5
- A.4 – Original Question (if question was MODIFIED from a BANK question)
- A.5 – Supplied Reference

B.0 – Regulator Documents
- B.0 – Tech Specs
- B.1 – Tech Spec Bases
- B.2 – TRM
- B.3 – COLR
- B.4 – PTLR
- B.5 – ODCM
- B.6 – PLS
- B.7 – PTDB
- B.8 - FSAR

C.0 – Procedures
- C.1 – EOP
- C.2 – AOP
- C.3 – SOP
- C.4 – UOP
- C.5 – ARP
- C.6 – Admin
- C.7 - Surveillance
- C.9 – Other

D.0 – Drawings
- D.1 – P&IDs
- D.2 – Oneline
- D.3 – Elementary
- D.4 – Logic
- D.5 - Other

E.0 – Misc Other
- E.1 – Photographs
- E.2 – Maps
- E.3 – Rad Surveys
- E.4 – Lesson Plan
- E.5 - OE

1. 001K2.03 001/LOIT/RO/M/F 2.7/3.1/001K2.03/LO-TA-27008///

Given the following:

- Unit 1 is at 100% reactor power.

Which one of the following completes the following statement?

The Rod Control Logic Cabinet is energized from __(1)__ or __(2)__.

	__(1)__	__(2)__
A.	Vital 125 VDC Panel (1AD11)	Vital 120 VAC Panel (1AY1A)
B.	Vital 125 VDC Panel (1AD11)	Regulated 120 VAC Instrument Panel (1NYS)
C.	1A MG Set via a 120 VAC transformer	Vital 120 VAC Panel (1AY1A)
D.✓	1A MG Set via a 120 VAC transformer	Regulated 120 VAC Instrument Panel (1NYS)

K/A

001 Control Rod Drive System

K2.03 Knowledge of bus power supplies to the following:

- One-line diagram of power supplies to logic circuits.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the two power supplies to the Rod Control Logic and Power cabinets.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate must recall where the power source originates for the power supplies in the Rod Control Logic and Power cabinets. The source is the same for both cabinets, however the power supply output voltages are different from the cabinets. The primary source comes from the output of the MG sets. A single phase of the 260 VAC MG Set output is transformed and regulated to 120 VAC, which splits and feeds both cabinets. The second source comes from the 120 VAC Regulated Instrument Bus NYS, breaker 27, which also splits and feeds both cabinets. Reference one-line drawing 1X3D-AA-G05A and lesson plan simplified one-line V-LO-PP-27101, slide 32.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The Logic and Power cabinets are powered from the MG Sets and 1NYS-27 only. However, AD11 is a common 125 VDC feed to many safety-related components and bistables. AD11 is the control power supply to the 'A' RTBs. Additionally, the Rod Control MG Set Controllers are powered from non-1E 125 VDC. A candidate without specific knowledge of the rod control system could find it reasonable for the cabinets to be powered from the same source as the MG Set controllers or the reactor trip breakers.

The second part is incorrect. The Logic and Power cabinets are powered from the MG Sets and 1NYS-27 only. However, AY1A is the primary power feed to the SSPS logic bays. A candidate without specific knowledge of the rod control system could find it reasonable for the cabinets to also be powered from the 120 VAC Vital panel which is fed from AD1, especially if the candidate knows that rod control utilizes both DC and AC power.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. The Logic and Power cabinets are powered from the MG Sets and 1NYS-27.

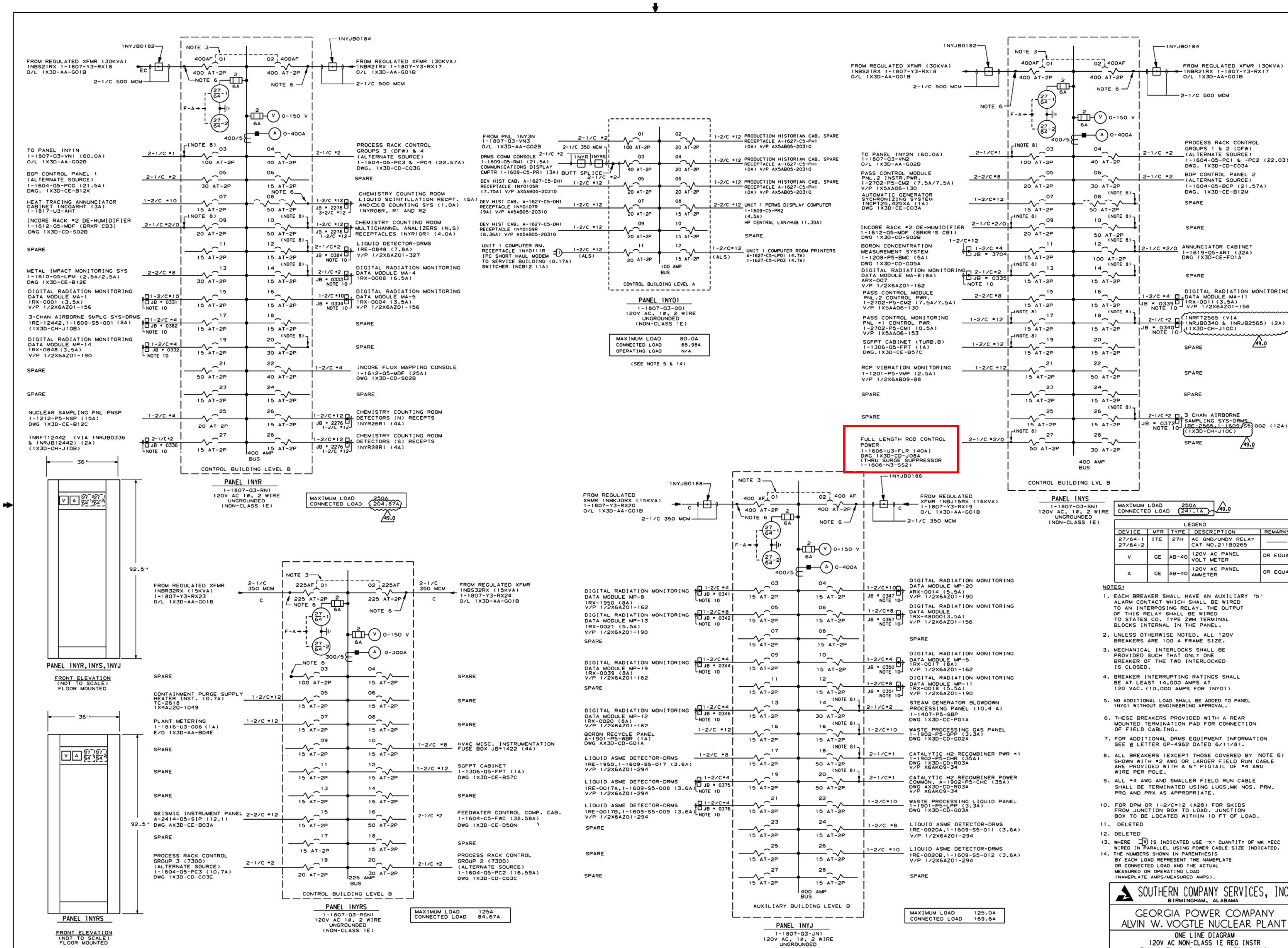
C. Incorrect. Plausible. The first part is correct. The Logic and Power cabinets are powered from the MG Sets and 1NYS-27.

The second part is incorrect. The Logic and Power cabinets are powered from the MG Sets and 1NYS-27 only. However, AY1A is the primary power feed to the SSPS logic bays. A candidate without specific knowledge of the rod control system could find it reasonable for the cabinets to also be powered from this common control related power supply.

D. Correct. Both parts are correct. The control and logic cabinets are powered from the MG set 260 VAC output via a 260/120 VAC transformer.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	001K2.03
Importance Rating:	2.7 / 3.1
Technical Reference:	V-LO-PP-27101 Rev 2.0 1X3D-AA-G05A Rev 49.0
References provided:	None
Learning Objective:	LO-PP-27101-02 State the power supplies for the Rod Control System. LO-TA-27008 Draw and label a one-line diagram of the Control Rod Drive Power Supply
Question origin:	BANK - Catawba 2007 Question #56
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.7
Comments:	This K/A was a replacement for 75K2.03 from the original sample plan.

You have completed the test!



LEGEND				
DEVICE	MFR	TYPE	DESCRIPTION	REMARKS
27/64-1 27/64-2	ITE	27H	AC GND/UNDV RELAY CAT NO.211B0265	———
V	GE	AB-40	120V AC PANEL VOLT METER	OR EQUAL
A	GE	AB-40	120V AC PANEL AMMETER	OR EQUAL

NOTES:

1. EACH BREAKER SHALL HAVE AN AUXILIARY 'b' ALARM CONTACT WHICH SHALL BE WIRED TO AN INTERPOSING RELAY. THE OUTPUT OF THIS RELAY SHALL BE WIRED TO STATES CO. TYPE ZWM TERMINAL BLOCKS INTERNAL IN THE PANEL.
2. UNLESS OTHERWISE NOTED, ALL 120V BREAKERS ARE 100 A FRAME SIZE.
3. MECHANICAL INTERLOCKS SHALL BE PROVIDED SUCH THAT ONLY ONE BREAKER OF THE TWO INTERLOCKED IS CLOSED.
4. BREAKER INTERRUPTING RATINGS SHALL BE AT LEAST 14,000 AMPS AT 120 VAC. (10,000 AMPS FOR 1NY01)
5. NO ADDITIONAL LOAD SHALL BE ADDED TO PANEL 1NY01 WITHOUT ENGINEERING APPROVAL.
6. THESE BREAKERS PROVIDED WITH A REAR MOUNTED TERMINATION PAD FOR CONNECTION OF FIELD CABLING.
7. FOR ADDITIONAL DRMS EQUIPMENT INFORMATION SEE W LETTER GP-4962 DATED 6/11/81.
8. ALL BREAKERS (EXCEPT THOSE COVERED BY NOTE 6) SHOWN WITH #2 AWG OR LARGER FIELD RUN CABLE ARE PROVIDED WITH A 6" PIGTAIL OF #4 AWG WIRE PER POLE.
9. ALL #4 AWG AND SMALLER FIELD RUN CABLE SHALL BE TERMINATED USING LUGS,MK NOS. PRM, PRQ AND PRX AS APPROPRIATE.
10. FOR DPM OR 1-2/C#12 (A28) FOR SKIDS FROM JUNCTION BOX TO LOAD. JUNCTION BOX TO BE LOCATED WITHIN 10 FT OF LOAD.
11. DELETED
12. DELETED
13. WHERE (X) IS INDICATED USE "X" QUANTITY OF MK #ECC WIRED IN PARALLEL USING POWER CABLE SIZE INDICATED.
14. THE NUMBERS SHOWN IN PARENTHESIS BY EACH LOAD REPRESENT THE NAMEPLATE OR CONNECTED LOAD AND THE ACTUAL MEASURED OR OPERATING LOAD (NAMEPLATE AMPS/MEASURED AMPS).

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

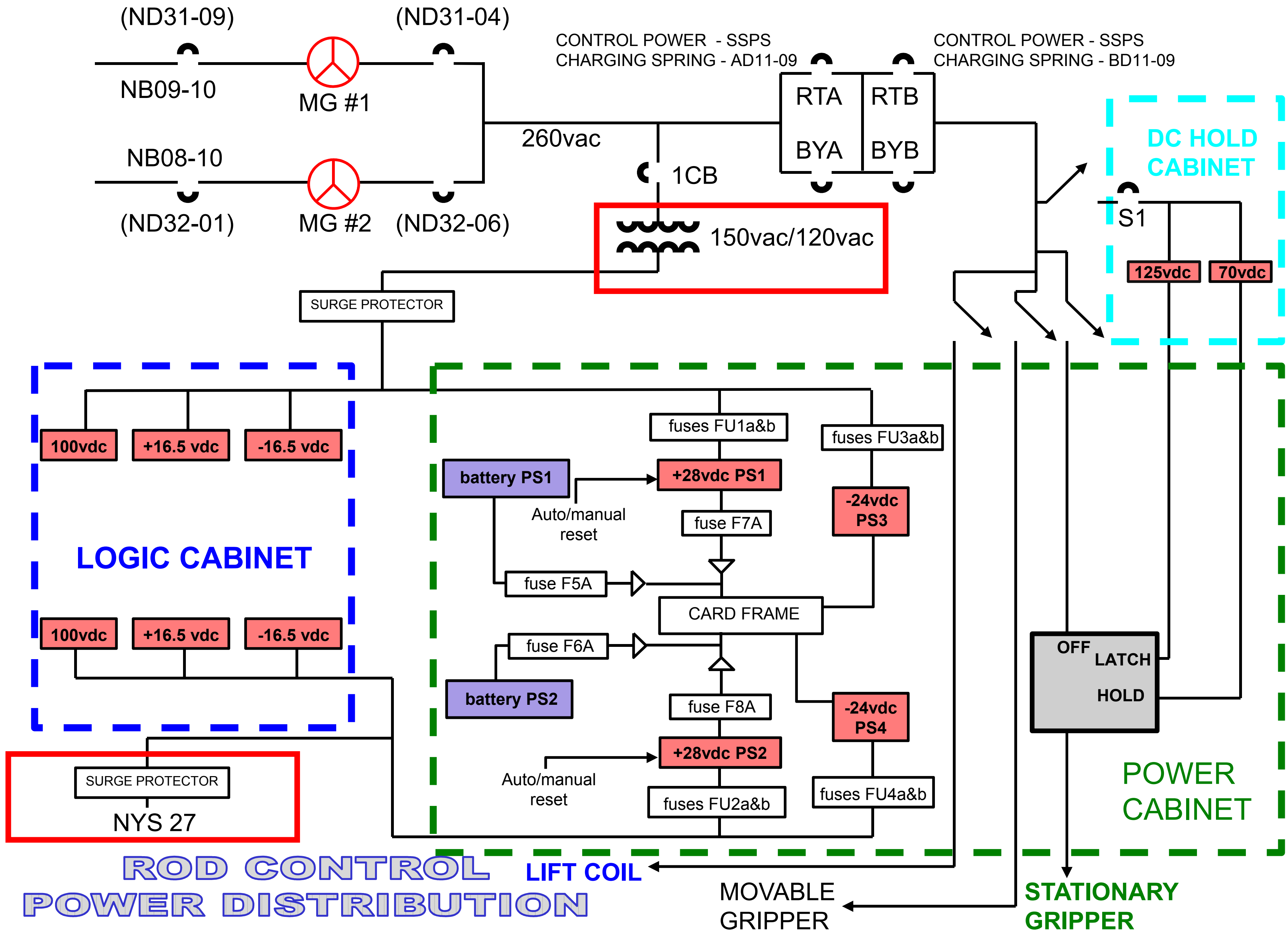
ONE LINE DIAGRAM
120V AC NON-CLASS 1E REG INSTR
PNL 1NYR, 1NYS, 1NYJ, 1NYRS & 1NY01
1-1807-Q3-RN1, SN1, JN1, RSN1, 001

NO.	VERSIONS	DATE	DR	CHK	APPV
49.0	REVISED PER ABN 1080131801E003, VER.2.0	04/10/13	TSL	GDM	JMR

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X3D-AA-G05A | VER. 49.0

SIZE E 34X44

DWG CATEGORY: CRITICAL



ROD CONTROL POWER DISTRIBUTION

1. 003A3.04 001/LOIT AND LOCT/RO/C/A 3.6/3.6/003A3.04/LO-PP-28103-03///

Initial condition:

- Unit 1 is at 8% reactor power.

Current condition:

- 1NAA de-energizes due to a fault on the bus.

Which one of the following completes the following statement?

When conditions stabilize, __(1)__ flow on 1FI-414, RCS Flow Loop 1 will be observed,

and

the reactor __(2)__ automatically trip due to the event.

	__(1)__	__(2)__
A.	~113%	will
B.	~113%	will NOT
C.	~6%	will
D.✓	~6%	will NOT

K/A

003 **Reactor Coolant Pump**

A3.04 **Ability to monitor automatic operation of the RCPS, including: RCS flow.**

K/A MATCH ANALYSIS

The question requires the candidate to monitor the status of the RCPs following undervoltage on a 13.8 kV bus and determine which RCS loops will still have flow.

EXPLANATION OF REQUIRED KNOWLEDGE

Above P-7 (10% NI), RCPs will trip on an **underfrequency** signal. The actuation of the undervoltage signal blocks the underfrequency relays to prevent all RCPs from tripping when voltage is lost on a single bus. Therefore, only two RCPs will be affected.

RCPs 1 and 3 are supplied by 13.8KV bus NAA, RCPs 2 and 4 are supplied by NAB.

Above P-7 and below P-8 (48% NI), loss of flow in two RCS loops will result in an automatic reactor trip. At 48% and above, a loss of flow in a single loop would result in

an automatic reactor trip.

RCPs normally indicate approximately 106% flow while in operation at NOPT. Once RCPs trip and reverse flow stabilizes, running RCPs indicate approximately 113% due to decreased backpressure and increased flow in the loops. The tripped RCPs indicate approximately 6% flow due to backflow in the loop.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. 1NAA feeds RCPs 1 & 3 and 6% flow would be observed. However, RCP power supplies are easily confused. If the candidate gets the two buses confused, RCPs 2&4 would be assumed to remain in operation and this choice would be correct.

The second part is incorrect. NI's are indicating less than the P-7 setpoint so the "at power trips" are not enabled and a loss of two RCPs will not result in an automatic reactor trip. However, if reactor power was greater than or equal to 10%, the reactor would trip.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. NI's are indicating less than the P-7 setpoint so the "at power trips" are not enabled and a loss of two RCPs will not result in an automatic reactor trip.

C. Incorrect. Plausible. The first part is correct. Bus 1NAA powers RCPs 1&3. The tripped RCPs indicate approximately 6% flow due to backflow in the loops.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	003A3.04
Importance Rating:	3.6 / 3.6
Technical Reference:	13503A-1 Rev 7.2 page 33 1X3D-AA-C01A Rev 23.0 1X6AA02-00229 Rev 1.0
References provided:	None
Learning Objective:	LO-PP-16401-09 - Describe the following for the RCP supply breakers: a. Breaker arrangement b. Power supply for each pump c. Protection features LO-PP-28103-03 - List all reactor trip set points, coincidences, permissives, and blocks.
Question origin:	NEW
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7 / 45.5
Comments:	None

You have completed the test!

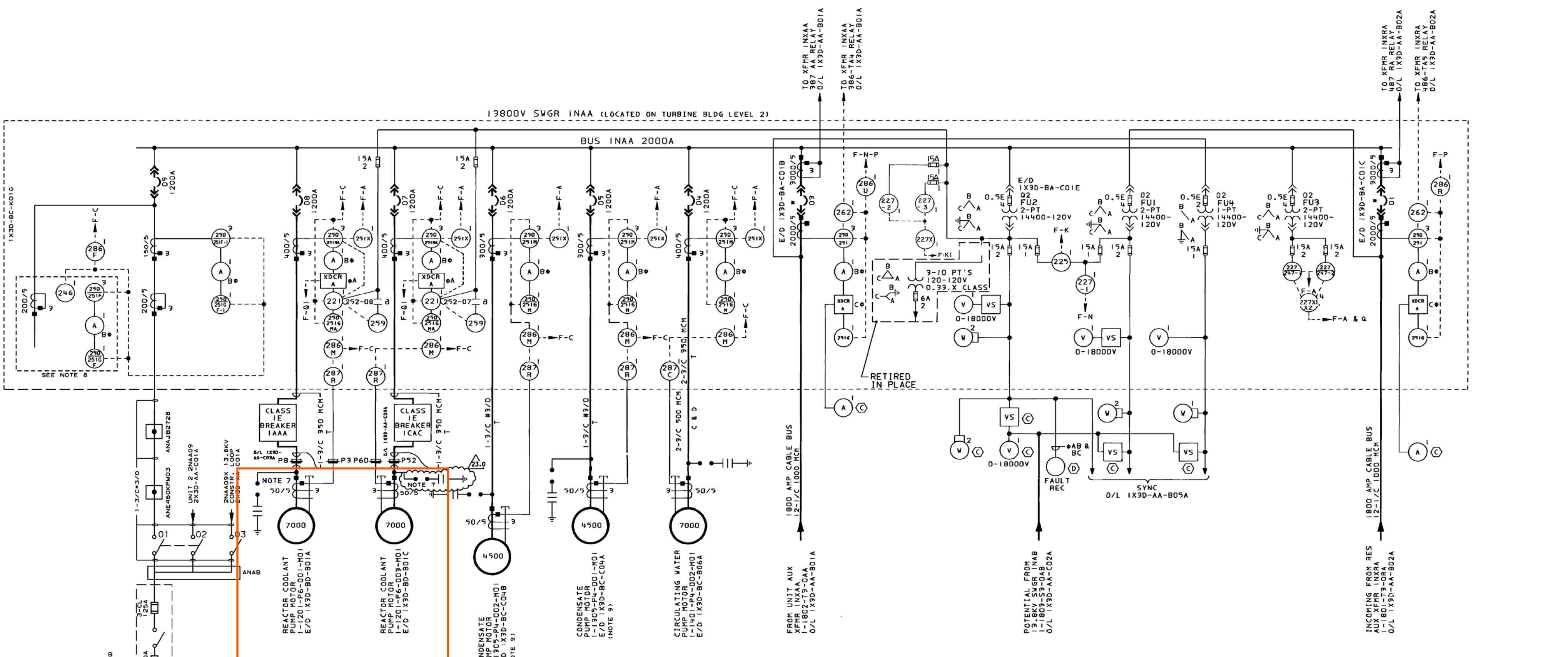
ATTACHMENT C

Sheet 1 of 6

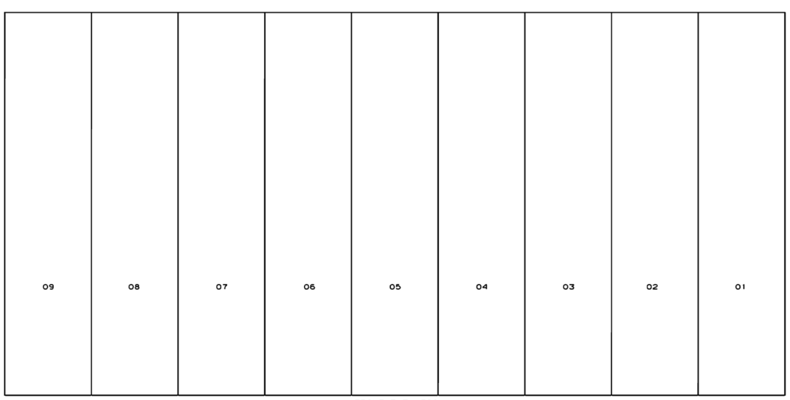
PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

PERMISSIVES

Permissive	Setpoint/Coincidence	Function
P-4	Train related Rx trip & Bypass breaker open	Trips Main Turbine Train A - mechanical Train B - electrical FWI if Lo Tavg (2/4 ≤ 564 °F) present Seals in FWI if caused by SI or P-14 (Hi Hi Level) Must be present to block auto SI after SI reset. P-4 Train A arms Steam Dumps P-4 Train B swaps Steam Dumps to plant trip controller
P-6	1/2 IR Detectors ≥ 2.0 E –5 % Rx Power	Allows manual block of SR High ϕ trip
P-7	P-10 (2/4 PR NIs ≥ 10% Rx power) or P-13 (PT-505 or 506 ≥ 10% turbine power)	Unblocks "At Power" Trips Przr Low Pressure Przr High Level RCS Two Loop Low Flow RCP UF RCP UV
P-8	2/4 PR NIS ≥ 48% Rx power	Enables Single Loop Low Flow Rx Trip
P-9	2/4 PR NIS ≥ 40% Rx power	Enables Turbine trip Rx trip
P-10	2/4 PR NIS ≥ 10% Rx power	Auto block of SR High ϕ trip Enables P-7 Allows manual block of IR rod stop and Hi ϕ trip Allows manual block of PR Hi ϕ trip Lo Setpoint
P-11	2/3 Przr Pressure channels ≤ 2000 psig	Auto enables Lo Przr Press SI & Lo Steamline Press SI/SLI & sends signal to open Accum Isolation Valves when P-11 resets. P-11 allows operator to block PRZR & Steamline low pressure SI & SLI. Also activates "Not Full Open" annunciators for Accumulator MOVs (ALB16; A5, B5, C5 & D5) and HV-8806; (ALB16 E03).
P-12	2/4 NR Tavg ≤ 550 °F	Interlocks Steam Dumps closed (Cooldown Dump Valves PV-507A, B & C) may be reopened by use of Bypass Interlock switches)
P-13	1/2 Turbine Impulse channels ≥ 10%	Enables P-7
P-14	2/4 NR SG Level channels ≥ 82%	Actuates FWI Actuates MFP and Main Turbine trip



FUNCTIONAL TABLE	
FUNCTION CODE - F	DESCRIPTION
A	ANNUNCIATOR
B	TRIPS ALL MOTOR FEEDER BREAKERS
C	TRIPS FEEDER BREAKER
K	PERMITS SUPPLY AUTO TRANSFER
N	BLOCKS SUPPLY AUTO TRANSFER
P	TRIPS SUPPLY FEEDER BREAKER
Q	COMPUTER LOG INPUT
R	OUTPUT TO PSC-A OR PSC-B
Q1	INPUT TO ERF
K1	PERMITS SUPPLY RESIDUAL TRANSFER



FRONT ELEVATION

13800V SWGR - 1NAA
SEE NOTE 1

LEGEND

DEVICE NUMBER	MFR	TYPE	MODEL NUMBER	DESCRIPTION
221	W	KD-10	ST#719B195A11	DISTANCE RELAY,30,0.75 - 21 OHM RANGE,FT 42 CASE
225	BECKWITH		M-0245B	SUPPLY TRANSFER SYNCHRONISM CHECK RELAY,120V
				60 HZ PHASE ANGLE LIMIT 0-60 DEGREE
227-1	GE	NGV	12NGV13A11A	SUPPLY TRANSFER UNDERVOLTAGE BLOCKING RELAY,
				70 - 100V
227X1-1,-2,227X2-1,-2	GE	HFA	12HFA51A42	UNDV/NEG SEQ AUX RELAY,125V DC
227/247-1-2	W	CVQ	ST#293B146A09	BUS UNDERVOLTAGE AND NEG SEQUENCE RLY
				55-140V,5 - 10% SENSITIVITY
246	GE	IJC	12IJC51B3A	CURRENT - BALANCE RLY,5A,1.0A MIN PICK-UP,
				2.5 SEC OPER TIME
250/251	GE	IAC	12IAC53M102A	BUS SUPPLY BREAKER PHASE OVERCURRENT RLY,
				VERY INVERSE,4-16A RANGE WITH HIGH DROPOUT
				INST UNIT 10 - 30A RANGE
251G	GE	IAC	12IAC53A801A	BUS SUPPLY BREAKER GND PROT RLY VERY INVERSE
				0.5 - 4.0A RANGE
250/251M	GE	IAC	12IAC66K37A	MOTOR PH OVERCURRENT RLY INVERSE LONG TIME 2-6A
				RANGE, 4-12A HIGH DROPOUT & 20-80A STD INST. UNITS
227X	GE	HGA	12HGA111J2	RESIDUAL XFER UNDERVOLTAGE AUX RELAY
250/251MA	GE	IFC	12IFC66KD3A	MOTOR PH OVERCURRENT RLY,INVERSE LONG TIME 2.5-7.5A
				RANGE,2-8A HIGH DROPOUT & 6-150A STD INST UNITS
250/251GM	GE	IAC	12IAC53B803A	MOTOR GND PROT RLY,VERY INVERSE,0.5 - 4.0A
				RANGE WITH 2-16A INST UNIT
250/251F	W	CO-9	ST#264C901A07	FEEDER PH OVERCURRENT RLY.
				VERY INVERSE,1-12A RANGE WITH INST UNIT 6-144A
250/251GF	W	CO-9	ST#264C901A07	FEEDER PH GND PROT RLY. VERY
				INVERSE,1-12A RANGE WITH INST UNIT 6-144A
251X	GE	NGA	12NGA15A24F	OVERCURRENT AUX ALARM RLY,WITH DIODE,125V DC
259	W	CV-4	ST#1875518	OVER VOLTAGE RELAY,55-140V RANGE,DPST,CC
262	W	TD-52	ST#606B354A09	TIMER,2 TIMERS PER CASE,0.05 - 0.4 & 0.05 - 1.0
				SEC RANGES 48/125V DC
286U,R	GE	HEA	12HEA61BRD235	SUPPLY BRKR AUX TRIPPING LOCKOUT RELAY,125V,
				5 OPEN & 5 CLOSED CONTACTS ON RESET POSITION
227-2-3	GE	GV	12NGV994M1A	SUPPLY RESIDUAL XFER UNDERVOLTAGE RELAY W/DROPOUT VLTG
				OF 30-50 VAC & WITH A PICKUP OF 20% OF DROPOUT SETTING
286M,F	GE	HEA	12HEA61ARD223	AUX TRIPPING LOCKOUT RELAY,125V,3 OPEN &
				3 CLOSED CONTACTS ON RESET POSITION
287R	BE	BE1-50	G2F-A1P-H1N6F	MOT DIFF PROT RLY,3 UNIT,0.25 - 2.0A RANGE
287C	GE	PJC	12PJC32K1A	MOT DIFF PROT RLY,3 UNIT,0.5 - 2.0A RANGE
				0.2/2.0A TARGET
A		LATER		AMMETER
V		LATER		VOLTMETER
VS		LATER		VOLTMETER SWITCH
W		LATER		INDICATING LIGHT - WHITE
XDCR-A	SC		CT-S10A2	TRANSDUCER,CURRENT,5A INPUT,1 MA OUTPUT,
				0.25% ACCURACY
250/251 F1	GE	IAC	12IAC53B812A	STA SVCE XFMR,FEEDER PHASE OVERCURRENT RLY,VERY INVS
				1.5-12.0A RANGE WITH 20-160A INST UNIT
250/251 GF-1	GE	IAC	12IAC53B803A	STA SVCE XFMR,FEEDER GND PROT RLY,VERY INVS 0.5-4.0A
				RANGE WITH 2-16A INST UNIT
250/251 GMA	GE	IFC	12IFC53BD2A	MTR GND PROT RLY,VERY INVS 0.5-4.0A RNG,2-50A INST UNIT

ABBREVIATIONS:

T - TRAY
C - CONDUIT (RIGID GALV. STEEL)
D - UNDERGROUND DUCT
• - 2000 AMP BREAKER

NOTES:

1. RELAY AND INSTRUMENTATION FRONT VIEW LAYOUT OF SWITCHGEAR 1NAA WILL BE SIM. TO SWITCHGEAR 1NAB.
2. FOR DEVICE TABLE, GENERAL NOTES, REFERENCE DWGS & LEGEND SEE O/L 1X3D-AA-A00B.
3. DELETED
4. THE LETTER INSIDE THE HEXAGON INDICATES REMOTE LOCATION OF DEVICE AS FOLLOWS:
   - C ELECTRICAL AUXILIARY BOARD
   - D PLANT FAULT RECORDER CABINET
5. ALL POWER CABLES ENTER SWITCHGEAR FROM THE TOP.
6. DELETED
7. SURGE PROTECTION ON REACTOR COOLANT PUMPS AS SHOWN ON 1X6AB09-00081.
8. RELAYS REMOVED FROM HOUSING, CT'S SHORTED. HOUSING, AMMETER, CT'S AND WIRING RETIRED IN PLACE. CT LEADS SPARED AT CT END ONLY.
9. INFRARED WINDOWS (HWK-075-C-L) INSTALLED ON MOTOR JUNCTION BOX.

NO.	VERSIONS	DATE	DR	CHK	APPV
23.0	REVISED PER ABN-V02588, VER. 1.0	3/31/11	DCP	ELC	JMR

NUCLEAR SAFETY RELATED

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

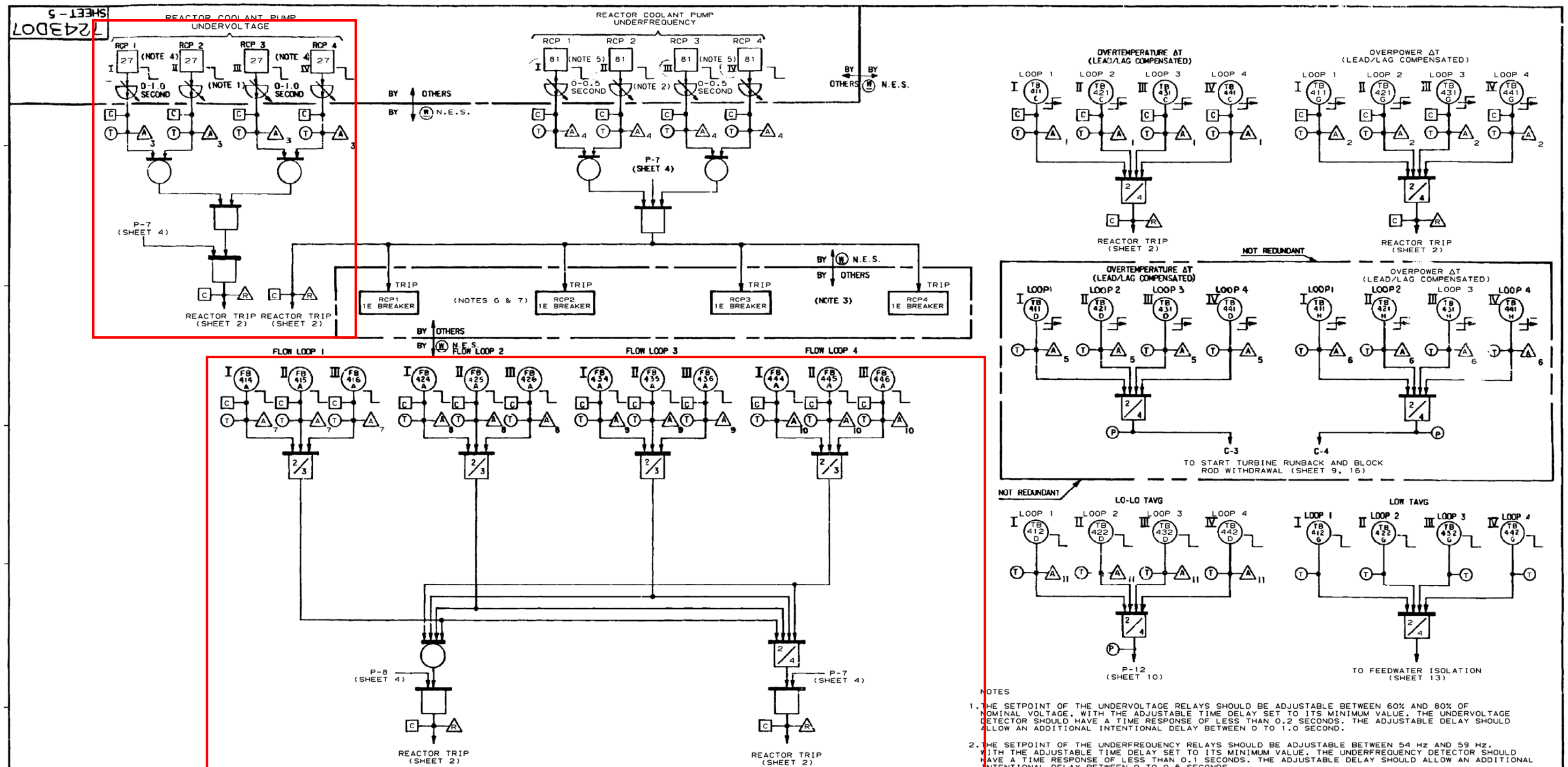
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ONE LINE DIAGRAM
13800V SWITCHGEAR 1NAA
1-1825-S3-0AA

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-C01A	23.0

SIZE E 34X44

DWG CATEGORY: CRITICAL



NOTES

1. THE SETPOINT OF THE UNDERVOLTAGE RELAYS SHOULD BE ADJUSTABLE BETWEEN 60% AND 80% OF NOMINAL VOLTAGE, WITH THE ADJUSTABLE TIME DELAY SET TO ITS MINIMUM VALUE. THE UNDERVOLTAGE DETECTOR SHOULD HAVE A TIME RESPONSE OF LESS THAN 0.2 SECONDS. THE ADJUSTABLE DELAY SHOULD ALLOW AN ADDITIONAL INTENTIONAL DELAY BETWEEN 0 TO 1.0 SECOND.
2. THE SETPOINT OF THE UNDERFREQUENCY RELAYS SHOULD BE ADJUSTABLE BETWEEN 54 Hz AND 59 Hz. WITH THE ADJUSTABLE TIME DELAY SET TO ITS MINIMUM VALUE. THE UNDERFREQUENCY DETECTOR SHOULD HAVE A TIME RESPONSE OF LESS THAN 0.1 SECONDS. THE ADJUSTABLE DELAY SHOULD ALLOW AN ADDITIONAL INTENTIONAL DELAY BETWEEN 0 TO 0.5 SECONDS.
3. THE MAXIMUM ALLOWABLE RCP BREAKER TRIP TIME DELAY IS 0.1 SECONDS.
4. THE UNDERVOLTAGE SENSORS (POTENTIAL TRANSFORMERS) MUST BE LOCATED ON THE MOTOR SIDE OF THE RCP CIRCUIT BREAKERS TO DETECT THE TRIP OF THE RCP CIRCUIT BREAKERS IN ADDITION TO BUS UNDERVOLTAGE.
5. THE UNDERFREQUENCY SENSORS MAY BE LOCATED ON THE MOTOR SIDE OF THE RCP CIRCUIT BREAKERS.
6. REACTOR COOLANT PUMPS NUMBER 1 AND 3 MUST BE ON BUS 1NAA. REACTOR COOLANT PUMPS NUMBER 2 AND 4 MUST BE ON BUS 1NAB. 11.0
7. REFER TO BECHTEL DRAWING SK-3-619 FOR DETAILED LOGIC FOR RCP BREAKERS.

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRIMARY COOLANT SYSTEM TRIP SIGNALS

SCALE ~ | 7243D07
DIMENSIONS IN INCHES | SHEET - 5
DO NOT SCALE | SUB 1656

CAD MICROSTATION LPM/03 1X6AA02-00229.DGN

THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
TAB/SECT. N/A
PAGE N/A
FIGURE N/A

VERSION 11.0 DATE 3-4-11
REVISED BY SNC PER ABN-V02456, VER. 1.0
SEE MICROFILM FOR PREVIOUS REV. SIGNATURES.

BY	CHK 'D	APPR.1	APPR.2
LPM	MLH	AAN	

Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT
UNIT NO. 1

TITLE:
FUNCTIONAL DIAGRAMS PRIMARY COOLANT SYSTEM TRIP SIGNALS

VENDOR: WESTINGHOUSE P.O.#:
DRAWING NO. 1X6AA02-00229
SIZE D

1. 004K6.09 001/LOIT/RO/C/A - 2.8/3.1/004K6.09/LO-PP-09100-03///

Initial conditions:

- Unit 1 is at 100% reactor power.

- VCT level is 50%.

- VCT makeup control is in AUTO.

Current condition:

- VCT level transmitter, 1LT-112, fails HIGH.

Assuming no operator action, which one of the following completes the following statements?

VCT Divert Valve, 1LV-112A, is aligned to the __(1)__.

If actual VCT level lowers to 29%, automatic makeup __(2)__ occur.

	__(1)__	__(2)__
A.	PRT	will
B.	PRT	will NOT
C.	RHUT	will
D.✓	RHUT	will NOT

K/A

004 **Chemical Volume Control System (CVCS):**

K6.09 **Knowledge of the effect of a loss or malfunction on the following CVCS components:**

- Purpose of VCT divert valve.

K/A MATCH ANALYSIS

The question addresses the effect of a failure of VCT Level Transmitter LT-112A (which results in the VCT divert valve aligning to the HUT), and the effect the transmitter failure will have on automatic VCT makeup.

EXPLANATION OF REQUIRED KNOWLEDGE

Letdown Divert valve LV-112A is controlled by both LT-185 and LT-112. Per ARP 17007-1 ALB07-E05, LV-112A will divert to the HUT if in AUTO and VCT level is >97%

on LT-112, or modulate to the HUT if LT-185 is greater than the setpoint on PIC-185 (normally 87%).

Per ARP 17007-1, annunciator ALB07-E05 automatic makeup starts when LT-112 indicates <30% and stops when LT-112 is >50%. With LT-112 failed failed high, automakeup will not start regardless of actual VCT level as sensed by LT-185. Furthermore, the automatic swap-over to the RWST suctions will not occur since only one transmitter will indicate less than 5.7%. CCPs will eventually cavitate without manual operator action.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per 17007-1, LT-112 >97% level will result in LV-112A tripping to the HUT position. However, all relief and divert valves in the letdown system are directed to either the PRT or the HUT based on location. A candidate with insufficient knowledge of the VCT level control circuit may assume that flow would be diverted to the PRT since this is the more common location within letdown.

The second part is incorrect. LT-0112 transmitter has failed high. Per ARP 17007-1, LT-112 will never indicate <30% and therefore auto makeup will not start. However, a candidate with insufficient knowledge of the VCT auto makeup control circuit may believe VCT makeup is controlled by LT-185, or both LT-185 and LT-112, and assume VCT makeup will not be affected.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. LT-0112 transmitter has failed high. Per ARP 17007-1, LT-112 will never indicate <30% and therefore auto makeup will not start.

C. Incorrect. Plausible. The first part is correct. Per 17007-1, LT-112 >97% level will result in LV-112A tripping open to the HUT position.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first choice of part C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	004K6.09
Importance Rating:	2.8 / 3.1

Technical Reference: 17007-1/2 Rev 29.1
1X4DB115 Rev 34.0
LO-PP-09100 CVCS Letdown (pages 113 and 129 in particular)

References provided: None

Learning Objective: LO-PP-09100-03 State the purpose and describe the control signals, setpoints, and any interlocks for the following:
b. VCT divert valve, LV-112A
LO-PP-09300-05 Describe the automatic control functions for the Reactor Makeup Control System components during:
a. automatic makeup to the VCT

Question origin: BANK

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.7

Comments:

You have completed the test!

WINDOW E05

ORIGIN	SETPOINT
1-LT-0112	92% HI
1-LT-0185	20% LO

VCT
HI/LO LEVEL

1.0 **PROBABLE CAUSE**

1. High level:

   a. Volume Control Tank (VCT)/Hold-up Tank (HUT) Divert Valve 1-LV-0112A malfunction OR aligned to the VCT position,

   b. VCT makeup greater than charging flow.

2. Low level:

   a. Makeup Control NOT in automatic,

   b. VCT/HUT Divert Valve 1-LV-0112A malfunction OR aligned to the Recycle Holdup Tank position,

   c. System leak.

WINDOW E05
(Continued)

2.0 AUTOMATIC ACTIONS

NOTE

VCT automatic makeup should have started at 30 percent or stopped at 50 percent. ☐

1. Letdown flow diverts to the HUT WHEN 1-HS-0112A is in AUTO with VCT high level of 97 percent.

2. Charging Pump suction auto swaps to the Refueling Water Storage Tank (RWST) upon a Lo-Lo VCT level of 5.7 percent.

3. A summary of instrument setpoints associated with the VCT levels include:

LI-0112	VCT LEVEL	LI-0185
Trip open 112A	97%	Modulate 112A full divert (if LIC-0185 pot @8.70)
Hi level alarm	92%	
112A Trip Open signal Resets	87%	112A starts to divert (if LIC-0185 pot @8.70)
Auto Makeup stops	50%	
Auto Makeup starts	30%	
Low level alarm	20%	Low level alarm
RWST auto swapover	5.7%(2 of 2)	RWST auto swapover

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** VCT level using 1-LI-0185 on the QMCB AND compare to 1-LI-0112 on the IPC OR on Trend Recorder XR-40053.

2. IF equipment failure is indicated by EITHER LT-0185 OR LT-0112 failed high, **perform** the following:

   a. **Place** 1HS-112A to the VCT position.

> NOTE
>
> Pump cavitation may be indicated by fluctuating discharge pressure and/or erratic flow. ☐

   b. **Monitor** charging pump(s) for signs of cavitation. IF cavitation is observed:

      (1) **Isolate** letdown,

      (2) **Stop** any running charging pumps,

      (3) **Initiate** 18007-C Section B.

   c. **Initiate** Manual VCT Makeup per 13009-C**.**

   d. **Contact** maintenance to initiate repairs.

3. IF level is low AND makeup is lost, **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

4. IF level is low due to system leakage, **initiate** 18004-C, ”Reactor Coolant System Leakage**.”**

WINDOW E05
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF VCT level is high:

   a. **Stop** Makeup,

   b. **Divert** letdown flow to the Recycle Holdup Tank (HUT position) using 1-HS-0112A on the QMCB,

   c. **Operate** makeup per 13009-1, "CVCS Reactor Makeup Control System."

2. IF equipment failure is indicated, **initiate** maintenance as required.

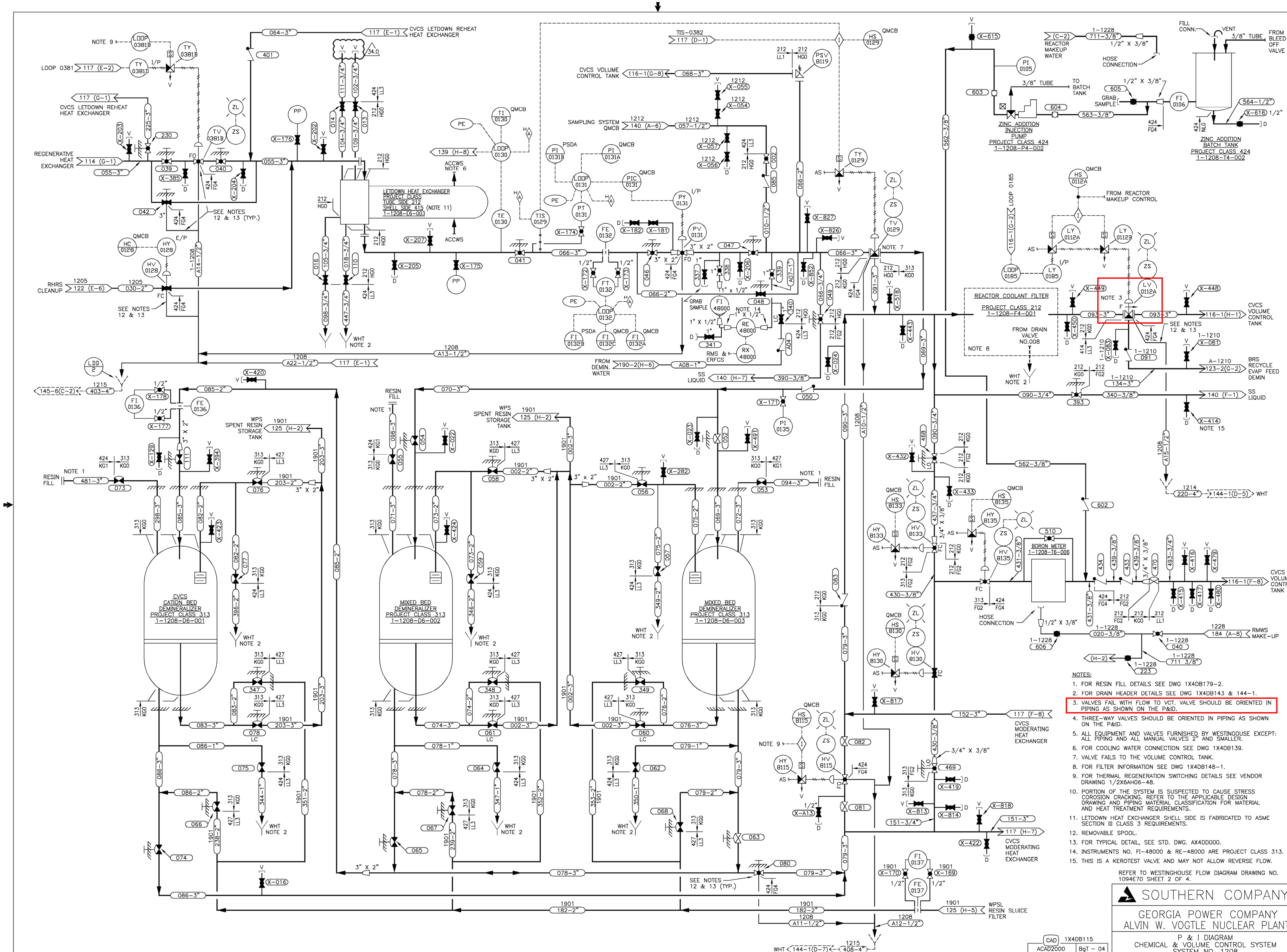
3. IF an operating charging pump fails due to suspected gas binding (fluctuating discharge pressure AND flow), THEN the standby pump SHALL NOT be started UNTIL the cause of the gas binding is understood AND all effected piping and components have been vented.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB115, 1X4DB116-1, 1X3D-BD-C02E, 1X6AU01-184, PLS, 1X5DT0012



NOTES:

1. FOR RESIN FILL DETAILS SEE DWG 1X4DB179-2.
2. FOR DRAIN HEADER DETAILS SEE DWG 1X4DB143 & 144-1.
3. VALVES FAIL WITH FLOW TO VCT. VALVE SHOULD BE ORIENTED IN PIPING AS SHOWN ON THE P&ID.
4. THREE-WAY VALVES SHOULD BE ORIENTED IN PIPING AS SHOWN ON THE P&ID.
5. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGOUSE EXCEPT: ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
6. FOR COOLING WATER CONNECTION SEE DWG 1X4DB139.
7. VALVE FAILS TO THE VOLUME CONTROL TANK.
8. FOR FILTER INFORMATION SEE DWG 1X4DB148-1.
9. FOR THERMAL REGENERATION SWITCHING DETAILS SEE VENDOR DRAWING 1/2X6AH06-48.
10. PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS COROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATION FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
11. LETDOWN HEAT EXCHANGER SHELL SIDE IS FABRICATED TO ASME SECTION III CLASS 3 REQUIREMENTS.
12. REMOVABLE SPOOL.
13. FOR TYPICAL DETAIL, SEE STD. DWG. AX4DD000.
14. INSTRUMENTS NO. FI-48000 & RE-48000 ARE PROJECT CLASS 313.
15. THIS IS A KEROTEST VALVE AND MAY NOT ALLOW REVERSE FLOW.

REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70 SHEET 2 OF 4.

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

CAD 1X4DB115 ACAD2000 BgT - 04

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
34.0	REVISED PER ABN-V01936, VER. 1.0	1-11-12	BgT	JWL	AAN	

SCALE: NONE
JOB NO.10604
DRAWING NO. 1X4DB115
VER. 34.0
SIZE E 34X44
DRAWING CATEGORY: CRITICAL

1. 005A2.02 001/LOIT AND LOCT/RO/C/A 3.5/3.7/005A2.02/LO-TA-60007///

Initial conditions:

- Unit 1 is in Mode 5 with solid plant conditions.
- RHR Train 'B' is in service aligned to low pressure letdown.
- RHR Train 'A' is in standby.
- CCP 'A' is in service.

Current condition:

- Instrument air header depressurizes due to an air line break.

Which one of the following completes the following statements?

With no operator action, RCS pressure will __(1)__.

Per 18028-C, "Loss of Instrument Air," to mitigate the pressure transient the crew will stop the __(2)__ pump.

	__(1)__	__(2)__
A.	increase	RHR
B.✓	increase	charging
C.	decrease	RHR
D.	decrease	charging

K/A

005 Residual Heat Removal

A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

Pressure transient protection during cold shutdown.

K/A MATCH ANALYSIS

(a) The question requires the candidate to predict the pressure transient caused by a malfunction of the RHR system from an instrument air loss during cold shutdown conditions with the primary "water solid".

(b) The question then requires the candidate to utilize Abnormal Operating Procedure (AOP) guidance to control/mitigate the resulting pressure transient.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to recall the failure modes of RHR letdown valves, RHR outlet and bypass valves, and Charging flow control valves.

The candidate is required to integrate failure modes against plant response, including contradictory responses from RHR and Charging.

The candidate is required to recall AOP response for loss of instrument air in lower modes with the plant in a solid condition.

Narrative of Response:
A loss of instrument air will cause RHR low pressure letdown flow control valve HV-128 to fail closed, resulting in a loss of letdown flow. Simultaneously, charging flow control valve FV-121 will fail open resulting in maximum charging flow (>150 gpm). In a solid plant condition, pressure will increase rapidly until the RHR suction reliefs lift at 450 psig. Per AOP 18028-C guidance on steps B3 and B4, all charging pumps will be stopped.

Additionally, the RHR outlet valve will fail open and the bypass valve will fail closed, resulting in maximum RHR flow with maximum cooling. In a solid plant condition, the pressure increase from the charging flow with no letdown has a far greater magnitude than the pressure reduction from the increased cooling (1 degree F is approximately 100 psig). In accordance with AOP18028-C guidance on steps B7 thru B9, the one running RHR pump will NOT be stopped, and instead an operator will be dispatched to the pump discharge valve to limit the cooldown.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. A loss of instrument air will result in RHR low pressure letdown flow control valve HV-128 failing closed, resulting in a loss of letdown flow. Simultaneously, charging flow control valve FV-121 will fail open resulting in maximum charging flow (>150 gpm). In a solid plant condition, pressure will increase rapidly until the RHR suction reliefs lift at 450 psig to control pressure

The second part is incorrect. Step B9 will not stop the RHR pump and instead will throttle the discharge flow. The distractor is plausible since the candidate may not consider the loss of letdown and increase in charging flow, and only consider the RHR flow and cooling increase. As such, 18028 step B9.b gives direction to stop an RHR pump.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per AOP 18028-C guidance on steps B3 and B4, if the RCS is solid, all charging pumps will be stopped.

C. Incorrect. Plausible. The first part is incorrect. The pressure increase caused by a

loss of letdown and max rate charging is significantly greater than the pressure reduction caused by an increase in RHR cooling. However, if the candidate does not consider the loss of letdown and max charging and only considers the increase in RHR flow and cooling, then RCS pressure would be expected to decrease due to density changes from the cooldown.

The second part is incorrect. AOP 18028-C does not give direction to stop the last RHR pump. 18028-C step B9.b. gives direction to stop <u>AN</u> RHR pump. To a novice operator, this direction may seem to be consistent with a pressure decrease caused by excessive cooling.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	005A2.02
Importance Rating:	3.5 / 3.7

Technical Reference: AOP 18028-C Rev 26.2
P&ID 1X4DB114 Rev 41.0
P&ID 1X4DB116-1 Rev 50.0
P&ID 1X4DB122 Rev 51.0

References provided: None

Learning Objective:

LO-LP-60321-08 Describe the effects on RCS pressure due to a loss of instrument air while solid on RHR.

LO-LP-60321-05 Describe why the RHR pump discharge should not be fully closed while throttling RHR flow to maintain RCS temperature during a loss of instrument air when in modes 4, 5, or 6.

LO-LP-60321-11 Given the entire AOP, describe:

a. Purpose of selected steps
b. How and why the step is being performed
c. Expected response of the plant/parameter(s) for the step

LO-TA-60007 Respond to a Loss of Instrument Air per 18028-C

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 43.5 / 45.3 / 45.13

Comments:

You have completed the test!

ATTACHMENT B

LOSS OF INSTRUMENT AIR IN MODES 4, 5, OR 6

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B1. Check Instrument Air supply header pressure on PI-9361 - LESS THAN 100 PSIG.	__B1. Go to Step B12.
__B2. IF a temporary air compressor is connected to the Service Air Header. Perform Attachment C while continuing with Attachment B.	

CAUTION

Loss of instrument air will cause CHARGING LINE CONTROL FV-0121 and SEAL FLOW CONTROL HV-0182 to fail open.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B3. Check RCS inventory – SOLID.	B3. Perform the following: __a. IF needed to maintain RCS level, THEN establish safety grade charging by initiating 13006, CHEMICAL AND VOLUME CONTROL SYSTEM. __b. Go to Step B6.
__B4. Trip all charging pumps.	
__*B5. **Monitor No. 1 seal leakoff temperature and flow until charging pump is restarted.**	

ATTACHMENT B

LOSS OF INSTRUMENT AIR IN MODES 4, 5, OR 6

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Loss of instrument air pressure will cause the RHR HX outlet valves to fail full open and the HX bypass valves to fail fully closed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__B6. Check plant Mode - MODE 4 OR MODE 5.	__B6. Suspend all fuel movement.
__B7. Check RCS temperatures – LOWERING.	B7. Perform the following: __a. Control temperature using ARVs. __b. IF ARVs NOT available, THEN stop all but one RCP.
__*B8. **Check RCS cooldown rate - GREATER THAN 100°F/HR.**	*B8. Perform the following: __a. Monitor cooldown rate. __b. IF cooldown rate can NOT be maintained less than 100°F/hr, THEN perform Step B9. __c. Go to Step B10.

LOSS OF INSTRUMENT AIR IN MODES 4, 5, OR 6

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE A key will be necessary to unlock 1205-U6-019 (Train A) or 1205-U6-020 (Train B).	
*B9. **Control RCS cooldown:**	
__a. Check RHR pump status - TWO PUMPS RUNNING.	__a. Go to Step B9.c.
__b. Stop one RHR pump.	
__c. Dispatch an operator to establish communications at the in service RHR heat exchanger inlet valve and local flow indicator:	

	UNIT 1	UNIT 2
TRAIN A	1-1204-U6-019 1-FIS-0610 (AB-C122)	2-1204-U6-019 2-FIS-0610 (AB-C38)
TRAIN B	1-1204-U6-020 1-FIS-0611 (AB-C92)	2-1204-U6-020 2-FIS-0611 (AB-C27)

Step 9 continued on next page

ATTACHMENT B

LOSS OF INSTRUMENT AIR IN MODES 4, 5, OR 6

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. Unlock and throttle 1205-U6-019 (Train A) or 1205-U6-020 (Train B) to maintain:	
__• RHR flow rate - GREATER THAN 750 GPM.	
__• RCS cooldown rate - LESS THAN 100°F/HR.	
__• CCW temperature at RHR HXs - LESS THAN 195°F. [ALB61-A01 NSCW CCW ACCW TRAIN A TEMP ALARM extinguished.] [ALB61-A02 NSCW CCW ACCW TRAIN B TEMP ALARM extinguished].	
__*B10. **Check Instrument Air header pressure - REMAINS GREATER THAN 70 PSIG.**	__*B10. Dispatch an operator to close Turbine Building Instrument Air isolation valve: UNIT 1: 1-2420-U4-512 (TB-1-TE12) UNIT 2: 2-2420-U4-512 (TB-1-TE10)
__B11. Check main turbine turning gear – ENGAGED.	__B11. Engage turning gear if necessary by initiating 13800, MAIN TURBINE OPERATION.

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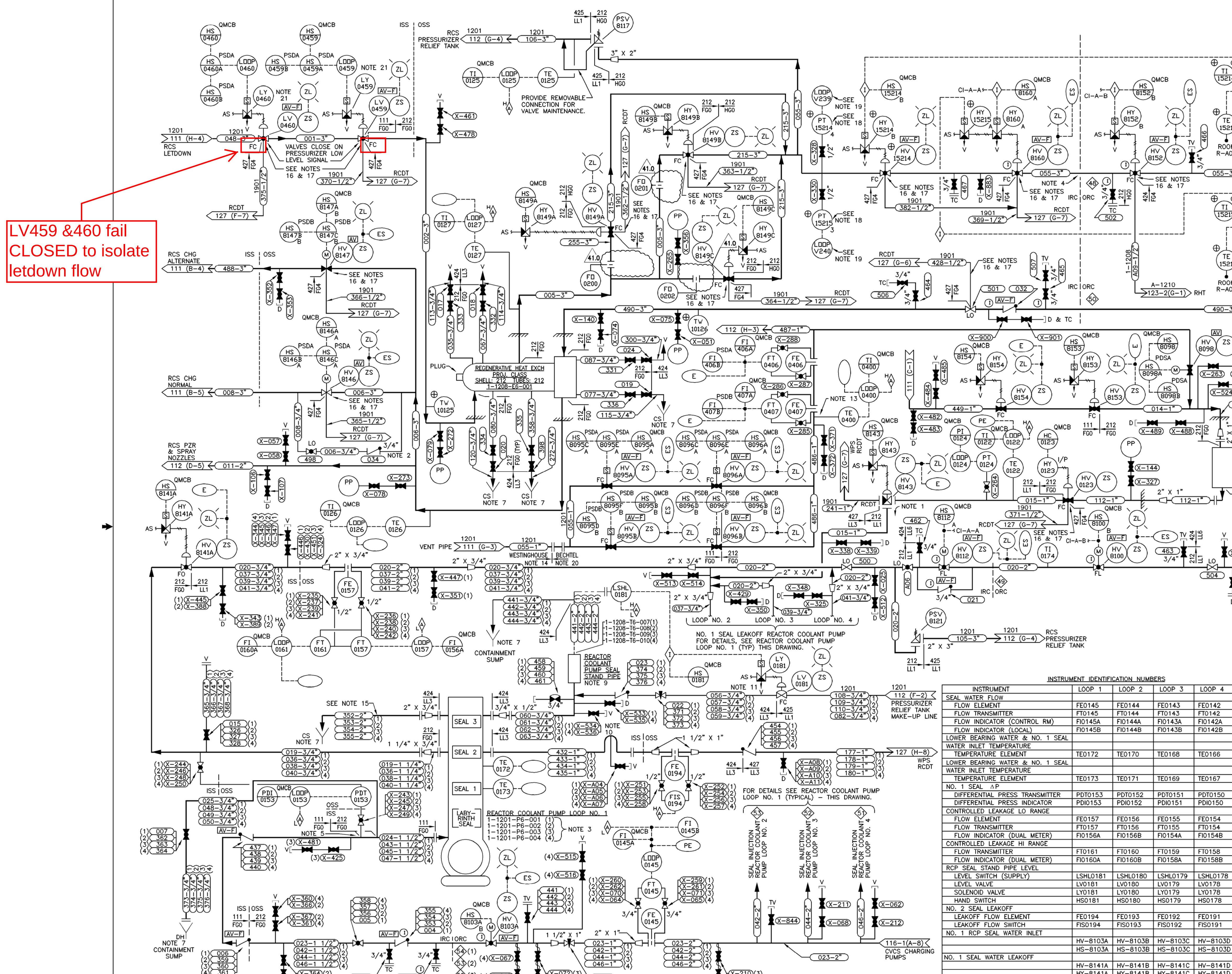


INSTRUMENT IDENTIFICATION NUMBERS

INSTRUMENT	LOOP 1	LOOP 2	LOOP 3	LOOP 4
SEAL WATER FLOW				
FLOW ELEMENT	FE0145	FE0144	FE0143	FE0142
FLOW TRANSMITTER	FT0145	FT0144	FT0143	FT0142
FLOW INDICATOR (CONTROL RM)	FI0145A	FI0144A	FI0143A	FI0142A
FLOW INDICATOR (LOCAL)	FI0145B	FI0144B	FI0143B	FI0142B
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0172	TE0170	TE0168	TE0166
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0173	TE0171	TE0169	TE0167
NO. 1 SEAL ΔP				
DIFFERENTIAL PRESS TRANSMITTER	PDT0153	PDT0152	PDT0151	PDT0150
DIFFERENTIAL PRESS INDICATOR	PDI0153	PDI0152	PDI0151	PDI0150
CONTROLLED LEAKAGE LO RANGE				
FLOW ELEMENT	FE0157	FE0156	FE0155	FE0154
FLOW TRANSMITTER	FT0157	FT0156	FT0155	FT0154
FLOW INDICATOR (DUAL METER)	FI0156A	FI0156B	FI0154A	FI0154B
CONTROLLED LEAKAGE HI RANGE				
FLOW TRANSMITTER	FT0161	FT0160	FT0159	FT0158
FLOW INDICATOR (DUAL METER)	FI0160A	FI0160B	FI0158A	FI0158B
RCP SEAL STAND PIPE LEVEL				
LEVEL SWITCH (SUPPLY)	LSHL0181	LSHL0180	LSHL0179	LSHL0178
LEVEL VALVE	LV0181	LV0180	LV0179	LV0178
SOLENOID VALVE	LY0181	LY0180	LY0179	LY0178
HAND SWITCH	HS0181	HS0180	HS0179	HS0178
NO. 2 SEAL LEAKOFF				
LEAKOFF FLOW ELEMENT	FE0194	FE0193	FE0192	FE0191
LEAKOFF FLOW SWITCH	FIS0194	FIS0193	FIS0192	FIS0191
NO. 1 RCP SEAL WATER INLET				
	HV-8103A	HV-8103B	HV-8103C	HV-8103D
	HS-8103A	HS-8103B	HS-8103C	HS-8103D
NO. 1 SEAL WATER LEAKOFF				
	HV-8141A	HV-8141B	HV-8141C	HV-8141D
	HY-8141A	HY-8141B	HY-8141C	HY-8141D
	HS-8141A	HS-8141B	HS-8141C	HS-8141D

NOTES:

1. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
2. SPRING LOADED CHECK VALVE.
3. NUMBERS IN PARENTHESIS BEHIND VALVE AND LINE BUBBLES, DESIGNATE LOOP NO.
   EXAMPLE: 001-1" (1) – LOOP NO. 1
   001-1" (2) – LOOP NO. 2
   REACTOR COOLANT PUMP SHOWN FOR LOOP NO. 1, TYPICAL FOR LOOPS 2 THRU 4, EXCEPT FOR INSTRUMENT AND VALVE IDENTIFICATION NUMBERS. SEE TABLE FOR INSTRUMENT IDENTIFICATION NUMBERS. RELATIVE POSITION OF VENTS AND DRAINS FOR LOOPS 2 THRU 4 MAY ALSO VARY.
4. DELAY TIME FROM REACTOR COOLANT SYSTEM TO THE CONTAINMENT PENETRATION MUST BE 45 SECONDS MINIMUM AT MAXIMUM LETDOWN FLOW.
5. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
6. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL 2" AND SMALLER MANUAL VALVES.
7. FOR DRAIN HEADER DETAILS SEE DWGS 1X4DB143, 144-1 & 144-2.
8. FOR AUX COMPONENT CLG WATER SYSTEM INFORMATION SEE DWG 1X4DB138-2.
9. BOTTOM OF STANDPIPE 31 TO 34 FEET ABOVE CONNECTION TO NO. 3 SEAL.
10. TOP OF LOOP TO RISE 6 INCHES ABOVE NO. 2 SEAL LEAKOFF CONNECTION; SLOPE PIPING DOWNWARD THROUGH RUN TO RCDT. LOCATE CHECK VALVE AT LEVEL OF RCDT.
11. VALVE SHOWN IS FOR LOOP ONE; VALVES IN OTHER LOOPS WILL USE THE APPROPRIATE RCP SEAL STANDPIPE INSTRUMENTATION FOR THAT LOOP.
12. A PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
13. CLAMPED TO PIPE SURFACE.
14. FURNISHED WITH REACTOR VESSEL HEAD.
15. SLOPE PIPING DOWNWARD FROM THIS POINT TO ASSURE 0 PSIG BACK PRESSURE ON THE NO. 3 SEAL.
16. PROVIDE 12" REMOVABLE SPOOL.
17. FOR TYPICAL DETAIL, SEE STD. DWG. AX4DD000.
18. INSTRUMENTS PT-15214 AND PT-15215 ARE TO BE ABANDONED IN PLACE. ELECTRICAL CONNECTION IS NOT REQUIRED.
19. LOOP DIAGRAMS 1X5DV239 AND 1X5DV240 ARE VOIDED. REFER TO WESTINGHOUSE FLOW DIAGRAMS, DRAWINGS NO. 1094E70 SHEET 1 OF 4, AND 1094E69 SHEET 2 OF 2.
20. VACUUM CONNECTION FOR REACTOR COOLANT VACUUM REFILL SYSTEM.
21. SOLENOIDS VALVES ARE PROJECT CLASSIFICATION 11J. SAFETY RELATED STATUS IS REQUIRED ONLY FOR THE SOLENOIDS TO ENSURE ACTIVE VALVE FUNCTION CAPABILITY OF ASSOCIATED AOV.

Q

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

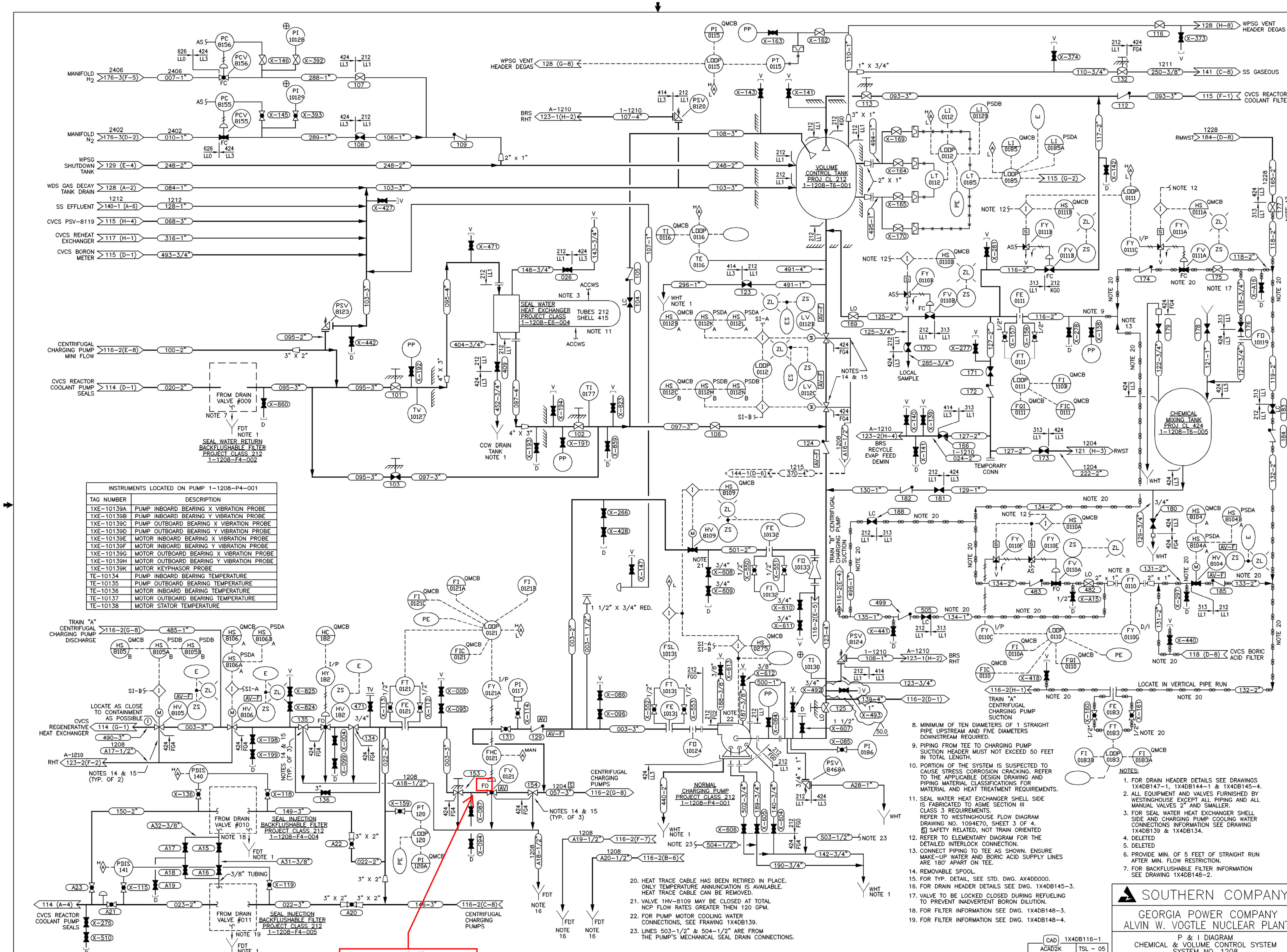
CAD 1X4DB114
ACAD2002 | TSL – 02

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
41.0	REVISED PER ABN 1090006501M002, VER. 1.0	10/07/09	TSL	MWD	JMR	X

SCALE:	DRAWING NO.	VER.
NONE	1X4DB114	41.0
JOB NO.10604		

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



INSTRUMENTS LOCATED ON PUMP 1-1208-P4-001	
TAG NUMBER	DESCRIPTION
1XE-10139A	PUMP INBOARD BEARING X VIBRATION PROBE
1XE-10139B	PUMP INBOARD BEARING Y VIBRATION PROBE
1XE-10139C	PUMP OUTBOARD BEARING X VIBRATION PROBE
1XE-10139D	PUMP OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139E	MOTOR INBOARD BEARING X VIBRATION PROBE
1XE-10139F	MOTOR INBOARD BEARING Y VIBRATION PROBE
1XE-10139G	MOTOR OUTBOARD BEARING X VIBRATION PROBE
1XE-10139H	MOTOR OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139K	MOTOR KEYPHASOR PROBE
TE-10134	PUMP INBOARD BEARING TEMPERATURE
TE-10135	PUMP OUTBOARD BEARING TEMPERATURE
TE-10136	MOTOR INBOARD BEARING TEMPERATURE
TE-10137	MOTOR OUTBOARD BEARING TEMPERATURE
TE-10138	MOTOR STATOR TEMPERATURE

NOTES:

1. FOR DRAIN HEADER DETAILS SEE DRAWINGS 1X4DB147-1, 1X4DB144-1 & 1X4DB145-4.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
3. FOR SEAL WATER HEAT EXCHANGER SHELL SIDE AND CHARGING PUMP COOLING WATER CONNECTIONS INFORMATION SEE DRAWING 1X4DB139 & 1X4DB134.
4. DELETED
5. DELETED
6. PROVIDE MIN. OF 5 FEET OF STRAIGHT RUN AFTER MIN. FLOW RESTRICTION.
7. FOR BACKFLUSHABLE FILTER INFORMATION SEE DRAWING 1X4DB148-2.
8. MINIMUM OF TEN DIAMETERS OF 1 STRAIGHT PIPE UPSTREAM AND FIVE DIAMETERS DOWNSTREAM REQUIRED.
9. PIPING FROM TEE TO CHARGING PUMP SUCTION HEADER MUST NOT EXCEED 50 FEET IN TOTAL LENGTH.
10. PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
11. SEAL WATER HEAT EXCHANGER SHELL SIDE IS FABRICATED TO ASME SECTION III CLASS 3 REQUIREMENTS.
    REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70, SHEET 3 OF 4.
    S SAFETY RELATED, NOT TRAIN ORIENTED
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. CONNECT PIPING TO TEE AS SHOWN. ENSURE MAKE-UP WATER AND BORIC ACID SUPPLY LINES ARE 180° APART ON TEE.
14. REMOVABLE SPOOL.
15. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
16. FOR DRAIN HEADER DETAILS SEE DWG. 1X4DB145-3.
17. VALVE TO BE LOCKED CLOSED DURING REFUELING TO PREVENT INADVERTENT BORON DILUTION.
18. FOR FILTER INFORMATION SEE DWG. 1X4DB148-3.
19. FOR FILTER INFORMATION SEE DWG. 1X4DB148-4.
20. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.
21. VALVE 1HV-8109 MAY BE CLOSED AT TOTAL NCP FLOW RATES GREATER THEN 120 GPM.
22. FOR PUMP MOTOR COOLING WATER CONNECTIONS, SEE FRAWING 1X4DB139.
23. LINES 503-1/2" & 504-1/2" ARE FROM THE PUMP'S MECHANICAL SEAL DRAIN CONNECTIONS.

CAD 1X4DB116-1
ACAD2K TSL - 05

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

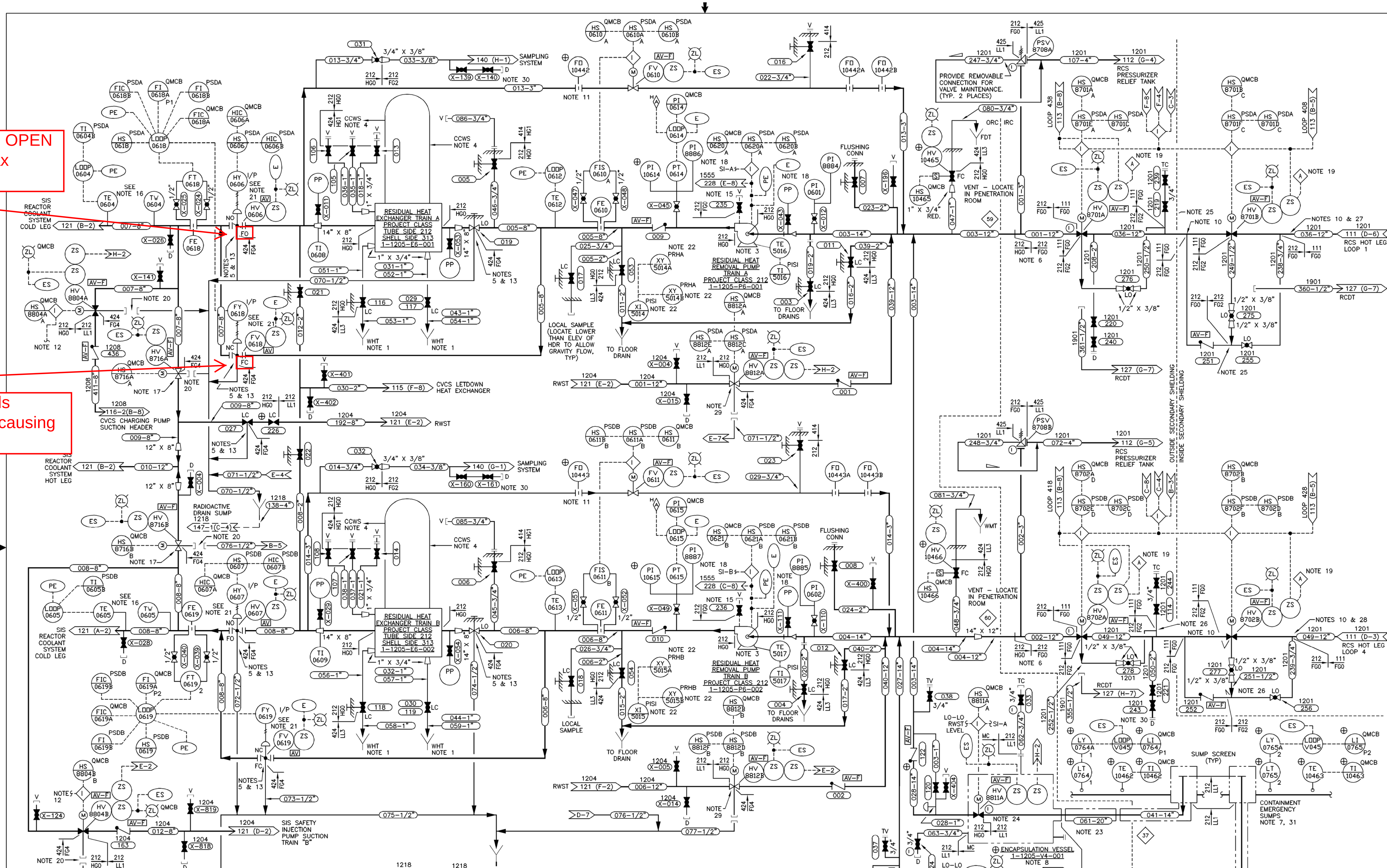
P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
50.0	REVISED PER ABN-1081944801M002, VER. 1.0	09/28/12	TSL	CFC	JMR	

SCALE: 0.0 | JOB NO.10604 | DRAWING NO. 1X4DB116-1 | VER. 50.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. FOR DRAIN HEADER INFORMATION SEE DWGS 1X4DB143, 144-1, 1X4DB145-5 AND 1X4DB183.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT: (1) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER, AND (2) EQUIPMENT MARKED ⊕.
3. FOR PUMP AND MOTOR COOLING WATER CONNECTIONS, SEE DWGS 1X4DB134 & 1X4DB137.
4. FOR HEAT EXCHANGER SHELL SIDE CONNECTIONS SEE DWG 1X4DB137.
5. PROVIDE 12" REMOVABLE SPOOL.
6. MAKE PIPING MATERIAL CLASS CHANGE AT FIRST WELD OF PENETRATION SLEEVE INSIDE THE CONTAINMENT.
7. INSTALL 150 #R.F. FLANGE ON INLET OF EACH 14"ø PIPE FOR PERIODIC TESTING. PROVIDE ONE MATING BLIND FLANGE TO BE DRILLED AND TAPPED FOR TEST CONNECTION. MATING FLANGE MUST BE REMOVED FOR NORMAL OPERATION.
8. VESSEL IS CONSTRUCTED TO SEISMIC CATEGORY 1 AND ASME III CLASS MC.
9. PORTIONS OF THIS SYSTEM ARE SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON DWG CX4DG001. 51.0
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. FOR TYPICAL DETAIL SEE STD. DWG AX4DD000.
14. DELETED.
15. HIGH POINT VENT VALVE (BECHTEL-FURNISHED) FOR SEAL PIPING - SEE 1X6AF02-25.
16. TE IS A SURFACE TYPE CLAMP ON RTD, TW IS INSTALLED BUT NOT USED.
17. VENT HOLE IS DRILLED ON RESPECTIVE RHR PUMP SIDE OF DISC.
18. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.
19. ALARM ON OPEN HOT LEG VALVE AND HIGH RCS PRESSURE.
20. DELETED.
21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.
23. 4" VIEW PORT TO BE USED TO VERIFY MOTOR OPERATED VALVE STROKE TEST WITHOUT REMOVING THE VESSEL HEAD.
24. VENT HOLE IS DRILLED ON CONTAINMENT SIDE OF DISC.
25. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-249-01.
26. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-251-01.
27. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-036-01.
28. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-049-02.
29. A SECTION OF THE LEAKOFF LINE HAS BEEN REMOVED AND BOTH ENDS CAPPED.
30. THESE ARE KEROTEST VALVES AND MAY NOT ALLOW REVERSE FLOW.
31. SUMP SCREEN TAG NO'S. 1-1205-G2-001 (TRAIN "A") AND 1-1205-G2-002 (TRAIN "B")

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. NO. 1094E72

CAD 1X4DB122
ACAD2007 | LPM - 07

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
RESIDUAL HEAT REMOVAL
SYSTEM NO. 1205

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
51.0	REVISED PER ABN-V02177, VERSION 1.0	05/19/10	LPM	LCF	PML	X

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB122	51.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 005AK3.05 001/LOIT/RO/M/F 3.4/4.2/005AK3.05/LO-TA-60037///

Initial conditions:

- Unit 1 is at 95% reactor power with a power ascension in progress.
- CBD is at 200 steps.

Current conditions:

- ALB10-D06 ROD DEV is received.
- CBD rod M12 DRPI indicates 109 steps.
- 18003-C, "Rod Control System Malfunction," is entered.
- Reactor power has been lowered per 18003-C guidance.

Which one of the following completes the following statement?

Per 18003-C, control rod M12 is considered __(1)__,

and

the reason for the power reduction is to minimize __(2)__ heat rates and power distribution variances.

	__(1)__	__(2)__
A.	dropped	core
B.	dropped	local
C.	misaligned	core
D.✓	misaligned	local

K/A

005 **Inoperable/Stuck Control Rod**

AK3.05 **Knowledge of the reasons for the following responses as they apply to the Inoperable / Stuck Control Rod:**

- Power limits on rod misalignment.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the reason power is reduced with a misaligned/dropped rod. The question also tests the candidate's knowledge of when a rod is considered dropped versus misaligned.

EXPLANATION OF REQUIRED KNOWLEDGE

Per AOP 18003-C Section A NOTE prior to step 1 and the section entry conditions, a rod misaligned greater than 110 steps is considered “dropped”.

Per AOP 18003-C Section A NOTE prior to step 11, power reduction as soon as practical after the rod drop occurs minimizes **local** fuel power distribution variances and the chances of fuel damage.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. A rod that is misaligned by more than 110 steps is considered dropped per 18003-C. Rod M12 is only misaligned 91 steps. However, rod M12 is only at a position of 109 steps. It is reasonable to assume a candidate who does not fully understand the NOTE in 18003-C may conclude that since the rod is <110 steps inserted into the core, the rod is considered dropped. Therefore, this distractor is plausible.

The second part is incorrect. When a rod is significantly misaligned, the power peaking that could result in fuel damage is a local area issue as opposed to an entire core issue. However, depending on location, a dropped or misaligned rod can significantly alter the core flux shape and drive AFD and QPTR outside limits. The flux issues seen over the core tend to be slower moving changes and are bounded by core design and Tech Spec limits. As such, these changes do not challenge fuel integrity. The localized flux changes are more significant and do challenge fuel integrity. A candidate with insufficient knowledge of core behavior on a dropped or misaligned rod could find it reasonable that core flux changes could damage fuel because Tech Spec limits could be approached.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. When a rod is misaligned, the peaking of power that could result in fuel damage is a local area issue as opposed to the entire core issue. This is described in the Note prior to step 11.

C. Incorrect. Plausible. The first part is correct. Rod M12 is only misaligned by 91 steps from its bank. Per the NOTE in 18003-C, the rod is considered "misaligned'.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G2
K/A#	005AK3.05
Importance Rating:	3.4 / 4.2

Technical Reference: AOP 18003-C Rev 26.4

References provided: None

Learning Objective:

LO-LP-60303-01 Describe how the following Rod Control System malfunctions could result in xenon oscillations in the core:

a. dropped rod
b. uncontrolled continuous rod motion
c. misaligned rod

LO-LP-60303-14 Describe how the retrieval of a misaligned rod can affect the power distribution limits of the core. Include a effects on why Reactor Engineering must be consulted if the rod has been misaligned for longer than one hour.

LO-LP-60303-16 Describe how the radial flux profile may be affected by a misaligned rod.

LO-TA-60037 Respond to a Misaligned Control Rod per 18003-C

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.5, 41.10, 45.6, 45.13

Comments:

You have completed the test!

A. DROPPED RODS IN MODE 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A10. Check Annunciator ALB10-F02, Power Range Hi Neutron Flx Rate Alert is clear. • TSLB-4 NI Hi rate bistables not illuminated.	A10. On the affected PR NI Drawer, reset the Positive Rate Trip as follows: a. Turn the RATE MODE Switch momentarily to RESET. b. Verify POSITIVE RATE TRIP Drawer Light is NOT lit. c. Check PR HI RATE bistable not illuminated on TSLB-4 • 8.1 – PR HI Q RATE NC 41U • 8.2 – PR HI Q RATE NC 42U • 8.3 – PR HI Q RATE NC 43U • 8.4 – PR HI Q RATE NC 44U

NOTE

Power reduction as soon as practical after the rod drop occurs minimizes local fuel power distribution variances and the chances of fuel damage. Although TS 3.1.4 requires power be reduced to less that 75% within 2 hours, a target of achieving reactor power level less than 75% in one hour meets the “as soon as practical” objective.

A11. Reduce Thermal Power to less than 75% within 1 hour from time of Rod drop using 12004-C POWER OPERATION.

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure provides instructions for malfunctions of the Rod Control System resulting in uncontrolled rod motion, dropped or misaligned rods.

SYMPTOMS
SECTION A, DROPPED RODS IN MODE 1

- ALB10-E5 ROD AT BOTTOM
- ALB10-F2 POWER RANGE HI NEUTRON FLX RATE ALERT
- ALB10-C2 POWER RANGE CHANNEL DEVIATION
- Rod bottom LED on digital rod position indication.
- Rod misaligned greater than 110 steps from demand position
- Tavg dropping.

SECTION B, UNCONTROLLED CONTINUOUS ROD MOTION IN ALL MODES

- Rod motion with invalid demand from the Automatic Rod Control System.
- Failure of rods to stop moving when the Rod Motion Switch is released.

SECTION C, MISALIGNED RODS IN MODE 1

- ALB10-C2 POWER RANGE CHANNEL DEVIATION
- ALB10-D2 POWER RANGE UP DET HI FLX DEV
- ALB10-E2 POWER RANGE LWR DET HI FLX DEV
- Failure of ALB10-C4 ROD BANK LO LIMIT or ALB10-D4 ROD BANK LO-LO LIMIT to reset during rod withdrawal.
- Rod misaligned greater than 12 steps and less than or equal to 110 steps from demand position.
- Quadrant power tilt ratio calculation exceeds 1.02.

A. DROPPED RODS IN MODE 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE A Rod misaligned greater than 110 steps should be considered "dropped" and this section performed.	
A1. Stop any turbine loading changes.	
A2. Check the following:	A2. Perform the following:
• DRPI - AVAILABLE.	a. Trip the Reactor and Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
• Only one Rod dropped by observing DRPI.	
A3. Check rod misaligned greater than 110 steps.	A3. Go to Section C MISALIGNED RODS IN MODE 1.
A4. Initiate TS 3.1.4.	
A5. Initiate The Continuous Actions Page.	
*A6. **Maintain Tavg at program by performing the following as appropriate:**	
• Adjust Turbine load.	
• Dilute or borate.	
• Use manual Rod control.	

C. MISALIGNED RODS IN MODE 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C1. Stop any turbine loading changes.	
C2. Check only one Rod – MISALIGNED.	C2. Trip the Reactor and Go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
C3. Check rod misaligned less than or equal to 110 steps.	C3. Go to Section A, DROPPED RODS IN MODE 1.
C4. Check misaligned Rod - MISALIGNED BY GREATER THAN 12 STEPS.	C4. Go to 13502, CONTROL ROD DRIVE AND POSITION INDICATION SYSTEM to address restoration of the misaligned rod.
C5. Initiate TS 3.1.4.	

NOTE

Power reduction as soon as practical after the rod drop occurs minimizes local fuel power distribution variances and the chances of fuel damage. Although TS 3.1.4 requires power be reduced to less that 75% within 2 hours, a target of achieving reactor power level less than 75% in one hour meets the “as soon as practical” objective.

C6. Reduce Thermal Power to less than 75% within 1 hour from time of discovery of Rod misalignment using 12004-C, POWER OPERATION (MODE 1).

1. 005K5.09 001/LOIT/RO/M/F 3.2/3.4/005K5.09/LO-LP-39213-01///

Initial conditions:

- Unit 1 is in Mode 6 for refueling.
- RHR pump 'B' is in service.
- RHR pump 'A' is tagged out.

Current conditions:

- RHR pump 'B' is stopped to place a fuel assembly in the vicinity of the RCS hot leg nozzle.

- Chemistry requests permission to make a chemical addition to the RCS.

Which one of the following completes the following statement?

Per Tech Spec 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," the Chemistry technician __(1)__ allowed to make the chemical addition to the RCS,

and

RHR pump 'B' can remain stopped for a MAXIMUM of __(2)__ hour(s) in an 8 hour period.

	__(1)__	__(2)__
A.	is	1
B.	is	4
C.✓	is NOT	1
D.	is NOT	4

K/A

005 **Residual Heat Removal**

K5.09 **Knowledge of the operational implications of the following concepts as they apply the RHRS:**

- Dilution and boration considerations.

K/A MATCH ANALYSIS

The RHR system is in service for shutdown cooling in Mode 6, and requires the candidate to evaluate proposed plant activities with regards to the Tech Spec

requirements. One of the items evaluated is the operational implication of a chemical addition (dilution activity) in the current plant configuration.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to recall the Limiting Condition of Operation for Residual Heat Removal (RHR) and Coolant Circulation - High Water Level and apply it to current plant configuration.

Only one train of RHR is required to be OPERABLE and in operation per Tech Spec 3.9.5. Additionally, per the note in the LCO, the train of RHR can be removed from operation for up to 1 hour per 8 hour period provided the RCS boron concentration is not reduced. The bases for this note specifically discusses stopping RHR to allow placement of fuel assemblies near the loops.

Chemical additions are made via a chemical addition pot, which is aligned to the suction of the charging pumps. Demin water from the RMWST is aligned to the pot and is used to mix and flush the chemicals. Therefore, any addition of chemicals to the RCS will result in a dilution.

The Tech Spec knowledge requirement is above the line and therefore is applicable to RO knowledge level. Additionally, ROs are expected to consider impacts to plant equipment and Tech Specs prior to altering system alignments. The specific Bases knowledge mentioned above is not required to answer the question, it is included in the explanation to justify the selection of the current conditions framed by the question for realism.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. A chemical addition utilizes RMWST water; a dilution (regardless of how insignificant) is being performed. Therefore, it would not be allowed. However, the RCS volume in this configuration is substantial. The chemical addition would produce a negligible change in boron concentration. A candidate not familiar with the specifics of the Tech Spec note could find it reasonable to make such a negligible change. Additionally, Tech Spec 3.4.8 allows chemical additions in Mode 5 provided Shutdown Margin is verified and HFASA is OPERABLE..

The second part is correct. Per 3.9.5 ‘Residual Heat Removal (RHR) and Coolant Circulation - High Water Level’ NOTE, the RHR Pump may be stopped for <1 hour in an 8 hour period

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per 3.9.5 ‘Residual Heat Removal (RHR) and Coolant Circulation - High Water Level’ the RHR Pump may be stopped for <1 hour in an 8 hour period. However, 4 hours is a completion time associated with TS 3.9.5 condition ‘A’.

C. Correct. The first part is correct. A chemical addition utilizes RMWST water; a dilution (regardless of how insignificant) is being performed. Therefore, it would not be allowed. Per LCO 3.9.5 NOTE, no activities are allowed that would reduce the RCS boron concentration.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 005K5.09
Importance Rating: 3.2 / 3.4

Technical Reference: Tech Spec 3.9.5
Tech Spec Bases 3.9.5
P&ID 1X4DB116-1 Rev 50.0

References provided: None

Learning Objective: LO-LP-39213-01 For any given item in section 3.9 of Tech Specs, be able to:
a. State the LCO.
b. State any one hour or less required actions.

LO-LP-39213-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:
a. Whether any Tech Spec LCOs of section 3.9 are exceeded.
b. The required actions for all section 3.9 LCOs.

LO-LP-39213-04 Describe the bases for any given Tech Spec in section 3.9.

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.5 / 45.7

Comments:

You have completed the test!

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation – High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

--NOTE--
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	In accordance with the Surveillance Frequency Control Program

BASES

APPLICABLE SAFETY ANALYSES (continued)	RHR and Coolant Circulation - High Water Level satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

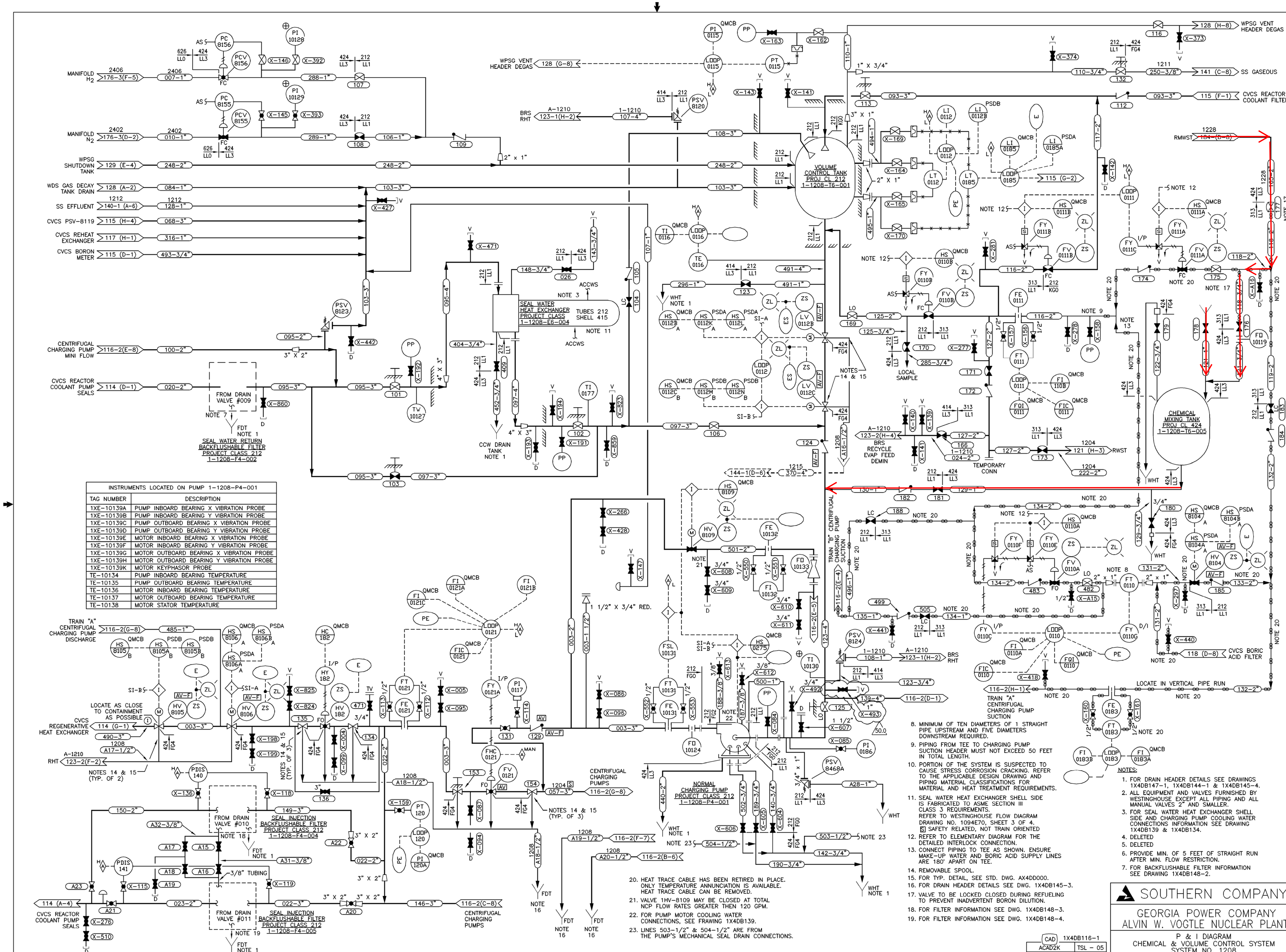
An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. The 23 ft water level was selected

(continued)



INSTRUMENTS LOCATED ON PUMP 1-1208-P4-001	
TAG NUMBER	DESCRIPTION
1XE-10139A	PUMP INBOARD BEARING X VIBRATION PROBE
1XE-10139B	PUMP INBOARD BEARING Y VIBRATION PROBE
1XE-10139C	PUMP OUTBOARD BEARING X VIBRATION PROBE
1XE-10139D	PUMP OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139E	MOTOR INBOARD BEARING X VIBRATION PROBE
1XE-10139F	MOTOR INBOARD BEARING Y VIBRATION PROBE
1XE-10139G	MOTOR OUTBOARD BEARING X VIBRATION PROBE
1XE-10139H	MOTOR OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139K	MOTOR KEYPHASOR PROBE
TE-10134	PUMP INBOARD BEARING TEMPERATURE
TE-10135	PUMP OUTBOARD BEARING TEMPERATURE
TE-10136	MOTOR INBOARD BEARING TEMPERATURE
TE-10137	MOTOR OUTBOARD BEARING TEMPERATURE
TE-10138	MOTOR STATOR TEMPERATURE

NOTES:

1. FOR DRAIN HEADER DETAILS SEE DRAWINGS 1X4DB147-1, 1X4DB144-1 & 1X4DB145-4.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
3. FOR SEAL WATER HEAT EXCHANGER SHELL SIDE AND CHARGING PUMP COOLING WATER CONNECTIONS INFORMATION SEE DRAWING 1X4DB139 & 1X4DB134.
4. DELETED
5. DELETED
6. PROVIDE MIN. OF 5 FEET OF STRAIGHT RUN AFTER MIN. FLOW RESTRICTION.
7. FOR BACKFLUSHABLE FILTER INFORMATION SEE DRAWING 1X4DB148-2.
8. MINIMUM OF TEN DIAMETERS OF 1 STRAIGHT PIPE UPSTREAM AND FIVE DIAMETERS DOWNSTREAM REQUIRED.
9. PIPING FROM TEE TO CHARGING PUMP SUCTION HEADER MUST NOT EXCEED 50 FEET IN TOTAL LENGTH.
10. PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
11. SEAL WATER HEAT EXCHANGER SHELL SIDE IS FABRICATED TO ASME SECTION III CLASS 3 REQUIREMENTS. REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70, SHEET 3 OF 4.
S SAFETY RELATED, NOT TRAIN ORIENTED
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. CONNECT PIPING TO TEE AS SHOWN. ENSURE MAKE-UP WATER AND BORIC ACID SUPPLY LINES ARE 180° APART ON TEE.
14. REMOVABLE SPOOL.
15. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
16. FOR DRAIN HEADER DETAILS SEE DWG. 1X4DB145-3.
17. VALVE TO BE LOCKED CLOSED DURING REFUELING TO PREVENT INADVERTENT BORON DILUTION.
18. FOR FILTER INFORMATION SEE DWG. 1X4DB148-3.
19. FOR FILTER INFORMATION SEE DWG. 1X4DB148-4.
20. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.
21. VALVE 1HV-8109 MAY BE CLOSED AT TOTAL NCP FLOW RATES GREATER THEN 120 GPM.
22. FOR PUMP MOTOR COOLING WATER CONNECTIONS, SEE FRAWING 1X4DB139.
23. LINES 503-1/2" & 504-1/2" ARE FROM THE PUMP'S MECHANICAL SEAL DRAIN CONNECTIONS.

CAD 1X4DB116-1
ACAD2K | TSL - 05

Q

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
50.0	REVISED PER ABN-1081944801M002, VER. 1.0	09/28/12	TSL	CFC	JMR	

SCALE: 0.0	DRAWING NO.	VER.
JOB NO.10604	1X4DB116-1	50.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 006A1.06 001/LOIT AND LOCT/RO/C/A 3.6/3.9/006A1.06/LO-TA-37008///

Initial conditions:

- Unit 1 experienced a small break LOCA.
- 19012-C, "Post-LOCA Cooldown and Depressurization," is in progress.
- Both RHR pumps are stopped.
- RCS temperature and pressure are stable.

Current condition:

- CCP 'A' is stopped per 19012-C.

Which one of the following completes the following statements?

When CCP 'A' is stopped, RCS subcooling margin will **initially** __(1)__.

Subsequently, if the OATC observes subcooling margin is 22°F and lowering after stopping SIP 'B', then the OATC is required to __(2)__ per 19012-C guidance.

	__(1)__	__(2)__
A.	increase	re-start SIP 'B'
B.	increase	re-actuate SI
C.✓	decrease	re-start SIP 'B'
D.	decrease	re-actuate SI

K/A

006 Emergency Core Cooling

A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including:

- Subcooling Margin

K/A MATCH ANALYSIS -

The question requires the candidate to predict the subcooling trend in response to stopping the first CCP during 19012-C, "Post-LOCA Cooldown and Depressurization". The candidate must determine the required actions when subcooling margin lowers below procedural limits.

EXPLANATION OF REQUIRED KNOWLEDGE

19012-C, "Post-LOCA Cooldown and Depressurization", establishes a 100 F/hour

cooldown and then secures ECCS pumps based on subcooling requirements. The question places the candidate in 19012-C after the completion of Step 25, when the CCP is stopped. The candidate must predict the effect of stopping the CCP (and subsequent pressure reduction) on subcooling.

There is a note just prior to Step 25 that reminds the operating crew to allow RCS pressure to stabilize or rise after an ECCS pump is secured before stopping another. The candidate is then asked to determine the correct action to mitigate the plant response. With subcooling <24F and lowering, ECCS flow must be increased. Step 36 gives the guidance to operate ECCS pumps as necessary and utilize Attachment C if needed to re-establish CCP Cold Leg Injection. The procedure then loops you back through to allow the cooldown to restore subcooling to a point that the ECCS pump can then be secured.

This response is contrasted to the guidance in other EOPs (ex. 19001-C) which direct the candidate to actuate SI if subcooling is <24F or Przr Lvl cannot be maintained >9%.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. RCS pressure will initially lower as the CCP is stopped, and subcooling will lower. However, a 100F/Hr cooldown rate is in progress priot to stopping the CCP, increasing subcooling as RCS temperature and pressure lower. The second part is correct. Per EOP 19012-C step 36 guidance, if subcooling continued to decrease to <24 F after plant condtions stabilized when the CCP was stopped, the RNO directs opperating ECCS pumps and re-establishing CCP Cold Leg Injection as necessary. Since SI 'B' was just stopped, it would therefore be re-started.

B. Incorrect. Plausible The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per step 36 and foldout page guidance, ECCS pumps are operated and Attachment C utilized to re-align to cold leg injection as necessary. The distractor is plausible since other EOPs direct the operator to actuate SI based on subcooling of <24F. Additionally, a candidate unfamiliar with the guidance of 19012-C could find it reasonable to re-actuate SI since the only actions that have occurred to this point were to stop one CCP and align through normal charging. Re-actuating SI would realign both of these without negatively affecting the remaining ECCS equipment.

C. Correct. The first part is correct. When the CCP is stopped, RCS pressure is expected to lower as injection flow is reduced. The reduction in RCS pressure results in a corresponding reduction in RCS subcooling. This is reflected in the NOTE prior to step 25 of EOP 19012-C.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible The first part is correct. See the first part of choice C above.

The second part is in correct. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 006A1.06
Importance Rating: 3.6 / 3.9

Technical Reference: 19012-C Rev 33.3

References provided: None

Learning Objective: LO-LP-37112-01 Using EOP 19012 as a guide, briefly describe how each step is accomplished.
LO-LP-37112-04 Given a NOTE or CAUTION statement from the EOP, state the bases for that NOTE or CAUTION statement.
LO-TA-37008 Perform Post-LOCA Cooldown and Depressurization of the RCS per 19012-C

Question origin: MODIFIED - LOIT Bank question LO-LP-37112-01-6

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 45.5

Comments:

You have completed the test!

1. LO-LP-37112-01 006/LOLP37112/LO-TA-13008/000EK3.06/////

A small break LOCA has occurred on Unit 1. The operating crew has transitioned to 19012-C, "ES-1.2 POST-LOCA COOLDOWN AND DEPRESSURIZATION", and are currently at the step which determines if a CCP may be stopped. The minimum subcooling requirements are met and the SS directs the OATC to stop CCP 1A.

Which **ONE** of the following parameters **CORRECTLY** describes the RCS response when Train 'A' CCP is stopped?

A. RCS pressure increases.

B.✓ RCS pressure decreases.

C. RCS break flow increases.

D. Pressurizer level increases.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*12. **Initiate RCS cooldown to cold shutdown:**	
a. Monitor shutdown margin by initiating 14005, SHUTDOWN MARGIN AND KEFF CALCULATIONS.	
b. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR.	Cooldown continuously increases subcooling, which allows ECCS pumps to be stopped sequentially.
c. Use RHR system if in service.	
d. Dump steam to Condenser from intact SG(s) using Steam Dumps:	d. Dump steam from intact SG(s) using SG ARV(s).
1) Place PIC-507 in Manual.	
2) Match demand on SG Header Pressure Controller PIC-507 and SD demand meter UI-500.	
3) Transfer Steam Dumps to STM PRESS mode.	
4) Open available Steam Dumps by slowly raising demand on PIC-507.	
13. Check RCS subcooling – GREATER THAN 24°F [38°F ADVERSE].	13. Go to Step 36.

°

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. PRZR Level - GREATER THAN 19% [50% ADVERSE].	c. Return to Step 15.
d. Start an RCP using ATTACHMENT A. (RCP 4 or RCP 1 preferred).	
e. Close PRZR Spray Valve(s) for stopped RCP(s): RCP 1: PIC-0455C RCP 4: PIC-0455B	

Reminder on pressure response from stopping ECCS pumps.

NOTES

- After stopping any ECCS Pump, RCS pressure should be allowed to stabilize or rise before stopping another ECCS pump.
- The CCPs and SI Pumps should be stopped on alternate ECCS trains when possible.

25. Check if one CCP should be stopped:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Two CCPs - RUNNING.	a. Go to Step 26.
b. Determine required RCS subcooling from table:	

SI Pump Status	Subcooling Criteria (°F)			
	With Any RCP Running		With No RCP Running	
	NORMAL	ADVERSE	NORMAL	ADVERSE
None Running	82	99	95	108
One Running	44	61	53	67
Two Running	40	57	49	63

° Step 25 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. RCS Subcooling - GREATER THAN REQUIRED SUBCOOLING.	c. IF RCS WR Hot Leg temperature greater than 350°F [340°F ADVERSE], THEN go to Step 36. IF RCS WR Hot Leg temperature less than 350°F [340°F ADVERSE], THEN perform the following: 1) Verify at least one RHR Pump running, 2) IF RHR Pump running, THEN go to Step 25.d. IF RHR Pumps can NOT be operated, THEN go to Step 36.
d. PRZR Level - GREATER THAN 19% [50% ADVERSE].	d. Return to Step 15.
Just after this step is where the question takes place.	
e. Stop one CCP.	
26. Check if one SI Pump should be stopped:	
a. Any SI Pump – RUNNING.	a. Go to Step 27.

Step 26 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Depressurize RCS until EITHER of the following conditions is satisfied: PRZR level - GREATER THAN 75% [52% ADVERSE]. -OR- RCS subcooling - LESS THAN 34°F [48°F ADVERSE].	
35. Verify adequate shutdown margin for xenon free cold shutdown by initiating 14005, SHUTDOWN MARGIN AND KEFF CALCULATIONS.	
36. Check ECCS flow not required:	
a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE]. Direction to re-established ECCS flow based on subcooling <24F. No	a. Perform the following: • Operate ECCS Pumps as necessary. • Initiate ATTACHMENT C as necessary to re-establish CCP Cold Leg Injection. • Go to Step 37.

Step 36 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. PRZR level - GREATER THAN 9% [37% ADVERSE].	b. Perform the following: • Operate ECCS Pumps as necessary. • Initiate ATTACHMENT C as necessary to re-establish CCP Cold Leg Injection. • Return to Step 15.
*37. **Check if SI Accumulators should be isolated:**	
a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE]	a. WHEN at least two RCS WR Hot Leg temperatures less than 380°F, THEN go to Step 38. IF at least two RCS WR Hot Leg temperatures NOT less than 380°F, THEN go to Step 39.
b. PRZR level - GREATER THAN 9% [37% ADVERSE].	b. Return to Step 15.

°

FOLDOUT PAGE

Guidance is also repeated on the foldout page.

1. SI REINITIATION CRITERIA
   Operate ECCS pumps as necessary if EITHER condition listed below occurs. Initiate ATTACHMENT C if it is necessary to re-establish CCP cold leg injection.

   - RCS subcooling - LESS THAN 24°F [38°F ADVERSE].

   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 9% [37% ADVERSE].

2. SECONDARY INTEGRITY CRITERIA
   Go to 19020-C, E-2 FAULTED STEAM GENERATOR ISOLATION, if any SG pressure is lowering in an uncontrolled manner or has been completely depressurized, and has not been isolated.

3. E-3 TRANSITION CRITERIA
   Go to 19030-C, E-3 STEAM GENERATOR TUBE RUPTURE, if any SG level rises in an uncontrolled manner or any SG has abnormal radiation.

4. COLD LEG RECIRCULATION SWITCHOVER CRITERION
   Go to 19013-C, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION, if RWST level lowers to less than 29%.

5. AFW SUPPLY SWITCHOVER CRITERION
   Switch to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM, when CST level lowers to less than 15%.

FOLDOUT

1. SI ACTUATION CRITERIA
   Actuate SI and go to Procedure 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION, if EITHER condition listed below occurs:

   - RCS subcooling – LESS THAN 24°F.

   - PRZR level – CANNOT BE MAINTAINED GREATER THAN 9%.

2. AFW SUPPLY SWITCHOVER CRITERION
   Switch to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM when CST level lowers to less than 15%.

3. Monitor SPENT FUEL POOL COOLING conditions:

   - Verify annunciators 17005-A6, SPENT FUEL PIT HI TEMP and 17005-E2, SPENT FUEL PIT LOW LEVEL are both clear.
     IF SPENT FUEL POOL LEVEL OR COOLING alarms are NOT clear,
     THEN initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING.

   - IF applicable, Using PTDB TAB 26, determine time to restore SFP LEVEL OR COOLING is < time to reach 200°F in Spent Fuel Pool.
     IF NOT initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING.

1. 006G2.4.47 001/LOIT AND LOCT/RO/C/A 4.2/4.2/006G2.4.47/LO-TA-37020///

At time 0100:

- Unit 1 reactor tripped.

At time 0530:

- Safety Injection occurred.
- All RCPs were stopped.

At time 0730:

- 19111-C, "Loss of Emergency Coolant Recirculation," is in progress.

- The crew is at Step 18, "Check if ECCS can be terminated," with the following conditions:

    - Containment pressure is 3.9 psig and stable.
    - Total ECCS flow is 625 gpm.
    - RCS WR pressure is 1045 psig and lowering.
    - CETC temperature is 475°F and lowering.
    - RVLIS Full Range is 65% and stable.

Which one of the following completes the following statement?

Based on the given conditions, the crew is required to _________ per 19111-C.

REFERENCE PROVIDED

A. terminate ECCS flow and establish normal charging

B✓ reduce ECCS flow to approximately 225 gpm

C. reduce ECCS flow to approximately 320 gpm

D. raise charging / ECCS flow as necessary

K/A

006 Emergency Core Cooling

G2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A MATCH ANALYSIS

The question requires the candidate to utilize plant parameters and trends during a Loss of Emergency Recirculation to diagnose ECCS effectiveness and determine the minimum amount of ECCS flow which is required based on guidance of 19111-C, Step

18, Table 1, and/or Figure 1 to conserve RWST inventory.

Subcooling is required to be calculated utilizing RCS WR Press, Core Exit Thermocouples, and steam tables. Minimum flow is determined by approximation on a graph or by linear interpolation from a table.

EXPLANATION OF REQUIRED KNOWLEDGE

Loss of Emergency Recirculation 19111-C checks for the ability to "terminate" SI in step 18. The EOP utilizes a relaxed termination criteria based on RVLIS Full Range and Subcooling. RVLIS requirements are modified by the number of RCPs running and subcooling is modified by the presence of adverse containment. These parameters determine the effectiveness of ECCS to cool the core. If sufficient cooling and inventory exist, SI is terminated and CCPs are aligned to the normal flow path. If sufficient inventory exist, but not sufficient subcooling, the EOP directs the operator to minimum ECCS flow in order to conserve RWST inventory. The minimum ECCS flow required is determined using either Table 1 or Figure 1 and is based on the decay heat, using time since Reactor Trip as the determining parameter. The flow determined is a minimum value and therefore the operator is expected to start and stop pump and operator valves within the control room as necessary to be at or above the minimum flow.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. RVLIS level is greater than required and subcooling is greater than the required non-adverse value. If the candidate does not recognize that adverse containment conditions exist, this would be the correct choice. Therefore, this distractor is plausible.

B. Correct. The candidate should determine that RVLIS level is greater than required, but subcooling is below the adverse value. This would require the candidate to determine a minimum flow from either the graph or table. The time since the event started is approximately 390 minutes which would result in a required flow of approximately 215 gpm, however this exact value is not given. The candidate must choose a flow rate greater than the required flow.

C. Incorrect. Plausible. The candidate should determine that RVLIS level is greater than required, but subcooling is below the adverse value. This would require the candidate to determine a minimum flow from either the graph or table. If the candidate incorrectly uses the time since SI reset instead of the time from reactor trip, 120 minutes would be used and a flow of approximately 315 gpm determined, however this exact value is not given. The candidate must choose a flow rate greater than the required flow.

D. Incorrect. Plausible. A common error candidates make is to navigate step 18 incorrectly. When evaluating RVLIS level, candidates go to the RNO when level is greater than specified instead of continuing down in the step. If this was occurred, the candidate would

continue to step 24 which would direct the operator to raise charging flow as necessary to maintain RVLIS level. Therefore, this distractor is plausible.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	006G2.4.47
Importance Rating:	4.2 / 4.2
Technical Reference:	19111-C Rev 33.2 WOG Background ECA-1.1 Rev 2, 4/30/2005
References provided:	19111-C Rev 33.2 step 18, Figure 1 and Table 1; pages 13-16, 45, and 46
Learning Objective:	LO-LP-37114-12 State the intent of EOP 19111, Loss of Emergency Coolant Recirculation. LO-PP-37115-02 Describe the actions taken to conserve RWST inventory for a loss of emergency coolant recirculation. LO-PP-37115-04 List the ECCS termination criteria and their bases for a loss of emergency coolant recirculation. LO-PP-37115-05 Discuss the ECCS injection flow control methods for a loss of emergency coolant recirculation. LO-PP-37115-07 Discuss parameters used to confirm adequate ECCS injection flow is being maintained during a loss of emergency coolant recirculation. LO-PP-37115-08 Desribe why the minimum required ECCS flow decreases with time following a reactor trip. LO-TA-37020 Respond to a Loss of Emergency Coolant Recirculation Capability per 19111-C
Question origin:	BANK - LOIT question HL-SR-00000-02-6
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.10 / 43.5 / 45.12
Comments:	

You have completed the test!

1. HL-SR-00000-02 006/HLSR00000/LO-TA-28004/061A2.04/////

A reactor trip with SI occurred at 0100. At 0600, the control room operators were directed to enter 19111-C, "Loss of Emergency Coolant Recirculation". The operators are now determining if SI flow can be terminated.

The following conditions currently exist at 0700:

- Total SI flow rate = 600 gpm
- CNMT Press = 3.8 psig
- RCS WR Pressure = 1035 psig
- CETC temperature = 500 degrees F
- All RCPs are OFF
- RVLIS Full Range = 65%

Based on these indications, the control room operators should:

A. Terminate SI and establish normal charging flow.

B.✔ Reduce total SI flow rate to approximately 175 gpm.

C. Reduce total SI flow rate to approximately 200 gpm.

D. Reduce total SI flow rate to approximately 540 gpm.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RCS Subcooling based on Core Exit TCs - GREATER THAN 24°F [38°F ADVERSE].	b. Close Spray Valve for idle RCP: RCP 1: PIC-0455C RCP 4: PIC-0455B Go to Step 18.
c. Start RCP 4 or RCP 1 or other RCP(s) as necessary to provide Normal Spray using ATTACHMENT D.	
d. Close Spray Valve for idle RCP: RCP 1: PIC-0455C RCP 4: PIC-0455B	
***18. Check if ECCS can be terminated:**	
a. Applicable RVLIS indication:	a. Go to Step 24.

RCP(s) running	Required Indication
0	Full Range greater than 63%
1	Dynamic Range greater than 25%
2	Dynamic Range greater than 34%
3	Dynamic Range greater than 50%
4	Dynamic Range greater than 72%

Step 18 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RCS Subcooling based on Core Exit TCs - GREATER THAN 74°F [88°F ADVERSE].	b. Establish minimum ECCS flow to remove decay heat by performing the following: 1) Determine minimum ECCS flow required using the following: TABLE 1 or FIGURE 1 2) Throttle ECCS flow to minimum value. 3) Go to Step 24.

CAUTION

Repositioning Phase A Isolation Valves may cause radiation problems throughout the plant.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. Reset Containment Isolation Phase A.	
20. Establish Instrument Air to Containment:	
a. Instrument Air pressure - GREATER THAN 100 PSIG.	a. Start additional Air Compressors as necessary.
b. Open INSTR AIR CNMT ISO VLV HV-9378.	
c. Verify PRZR Spray Valves operating as required.	

°

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. Stop ECCS Pumps and place in standby: • RHR Pumps • SI Pumps • All but one CCP	
22. Establish charging flow:	
a. Check Instrument Air - AVAILABLE.	a. Establish Safety Grade Charging by initiating 13006, CHEMICAL AND VOLUME CONTROL SYSTEM. Go to Step 23.
b. Verify CCP alternate miniflow valves in ENABLE PTL position: • HV-8508A - CCP-A RV TO RWST ISOLATION • HV-8508B - CCP-B RV TO RWST ISOLATION • Verify white Pressure Control Mode light - LIT	
c. Close BIT DISCH ISOLATION valves: • HV-8801A • HV-8801B	
d. Set SEAL FLOW CONTROL HC-182 to maximum seal flow (HV-0182 closed).	

Step 22 continued on next page

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

e. Open CHARGING TO RCS ISOLATION valves:

- HV-8105
- HV-8106

f. Establish desired charging flow using HV-0182 and FV-0121.

*23. **Maintain Seal Injection flow to all RCPs - 8 TO 13 GPM.**

*24. **Check adequate charging/ECCS flow:**

a. Applicable RVLIS indication:

RESPONSE NOT OBTAINED: a. Raise charging/ECCS flow to maintain RVLIS indication as necessary.

RCP(s) running	Required Indication
0	Full Range greater than 63%
1	Dynamic Range greater than 25%
2	Dynamic Range greater than 34%
3	Dynamic Range greater than 50%
4	Dynamic Range greater than 72%

b. Core Exit TCs - STABLE <u>OR</u> LOWERING.

RESPONSE NOT OBTAINED: b. Raise charging/ECCS flow to maintain TCs stable or lowering.

TABLE 1

Sheet 1 of 1

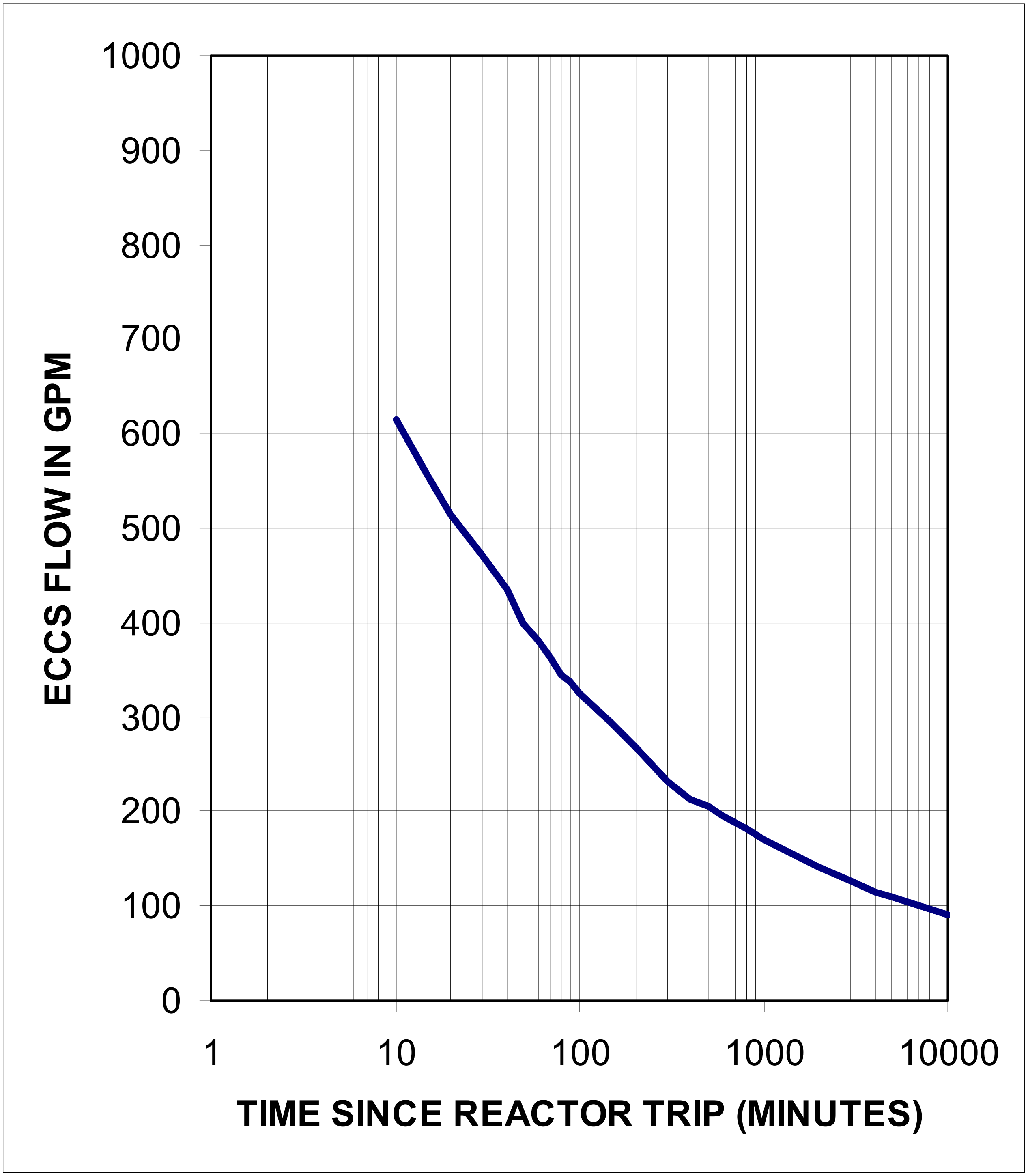
MINIMUM ECCS FLOW VERSUS TIME

Time Since Reactor Trip (Minutes)	ECCS Flow Rate (GPM)
10	615
15	555
20	515
30	471
40	435
50	399
60	381
70	363
80	344
90	337
100	326
150	294
200	268
300	232
400	214
500	205
600	196
800	181
1000	170
2000	141
3000	127
4000	116
5000	109
10000	91

FIGURE 1

Sheet 1 of 1

MINIMUM ECCS FLOW VERSUS TIME



ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RCS Subcooling based on Core Exit TCs - GREATER THAN 24°F [38°F ADVERSE].	b. Close Spray Valve for idle RCP: RCP 1: PIC-0455C RCP 4: PIC-0455B Go to Step 18.
c. Start RCP 4 or RCP 1 or other RCP(s) as necessary to provide Normal Spray using ATTACHMENT D.	
d. Close Spray Valve for idle RCP: RCP 1: PIC-0455C RCP 4: PIC-0455B	
***18. Check if ECCS can be terminated:**	
a. Applicable RVLIS indication:	a. Go to Step 24.

Distractor D

RCP(s) running	Required Indication
0	Full Range greater than 63%
1	Dynamic Range greater than 25%
2	Dynamic Range greater than 34%
3	Dynamic Range greater than 50%
4	Dynamic Range greater than 72%

65%

YES

Step 18 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. RCS Subcooling based on Core Exit TCs - GREATER THAN 74°F [88°F ADVERSE].	b. Establish minimum ECCS flow to remove decay heat by performing the following:
	1) Determine minimum ECCS flow required using the following:
	TABLE 1 or FIGURE 1
	2) Throttle ECCS flow to minimum value.
	3) Go to Step 24.

NO

Containment Pressure => Adverse values used.

1045 psig => 1060psia => 551.8 F Tsat => 76 F subcooling

CAUTION

Repositioning Phase A Isolation Valves may cause radiation problems throughout the plant.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. Reset Containment Isolation Phase A.	
20. Establish Instrument Air to Containment:	
a. Instrument Air pressure - GREATER THAN 100 PSIG.	a. Start additional Air Compressors as necessary.
b. Open INSTR AIR CNMT ISO VLV HV-9378.	
c. Verify PRZR Spray Valves operating as required.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. Stop ECCS Pumps and place in standby: • RHR Pumps • SI Pumps • All but one CCP	
22. Establish charging flow:	
a. Check Instrument Air - AVAILABLE.	a. Establish Safety Grade Charging by initiating 13006, CHEMICAL AND VOLUME CONTROL SYSTEM. Go to Step 23.
b. Verify CCP alternate miniflow valves in ENABLE PTL position: • HV-8508A - CCP-A RV TO RWST ISOLATION • HV-8508B - CCP-B RV TO RWST ISOLATION • Verify white Pressure Control Mode light - LIT	
c. Close BIT DISCH ISOLATION valves: • HV-8801A • HV-8801B	
d. Set SEAL FLOW CONTROL HC-182 to maximum seal flow (HV-0182 closed).	

Step 22 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

e. Open CHARGING TO RCS ISOLATION valves:

- HV-8105
- HV-8106

f. Establish desired charging flow using HV-0182 and FV-0121.

*23. **Maintain Seal Injection flow to all RCPs - 8 TO 13 GPM.**

*24. **Check adequate charging/ECCS flow:**

a. Applicable RVLIS indication:

Distractor D

RNO: a. Raise charging/ECCS flow to maintain RVLIS indication as necessary.

RCP(s) running	Required Indication
0	Full Range greater than 63%
1	Dynamic Range greater than 25%
2	Dynamic Range greater than 34%
3	Dynamic Range greater than 50%
4	Dynamic Range greater than 72%

65%

b. Core Exit TCs - STABLE OR LOWERING.

RNO: b. Raise charging/ECCS flow to maintain TCs stable or lowering.

TABLE 1

Sheet 1 of 1

MINIMUM ECCS FLOW VERSUS TIME

Time Since Reactor Trip (Minutes)	ECCS Flow Rate (GPM)
10	615
15	555
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30	471
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70	363
80	344
90	337
100	326
150	294
200	268
300	232
400	214
500	205
600	196
800	181
1000	170
2000	141
3000	127
4000	116
5000	109
10000	91

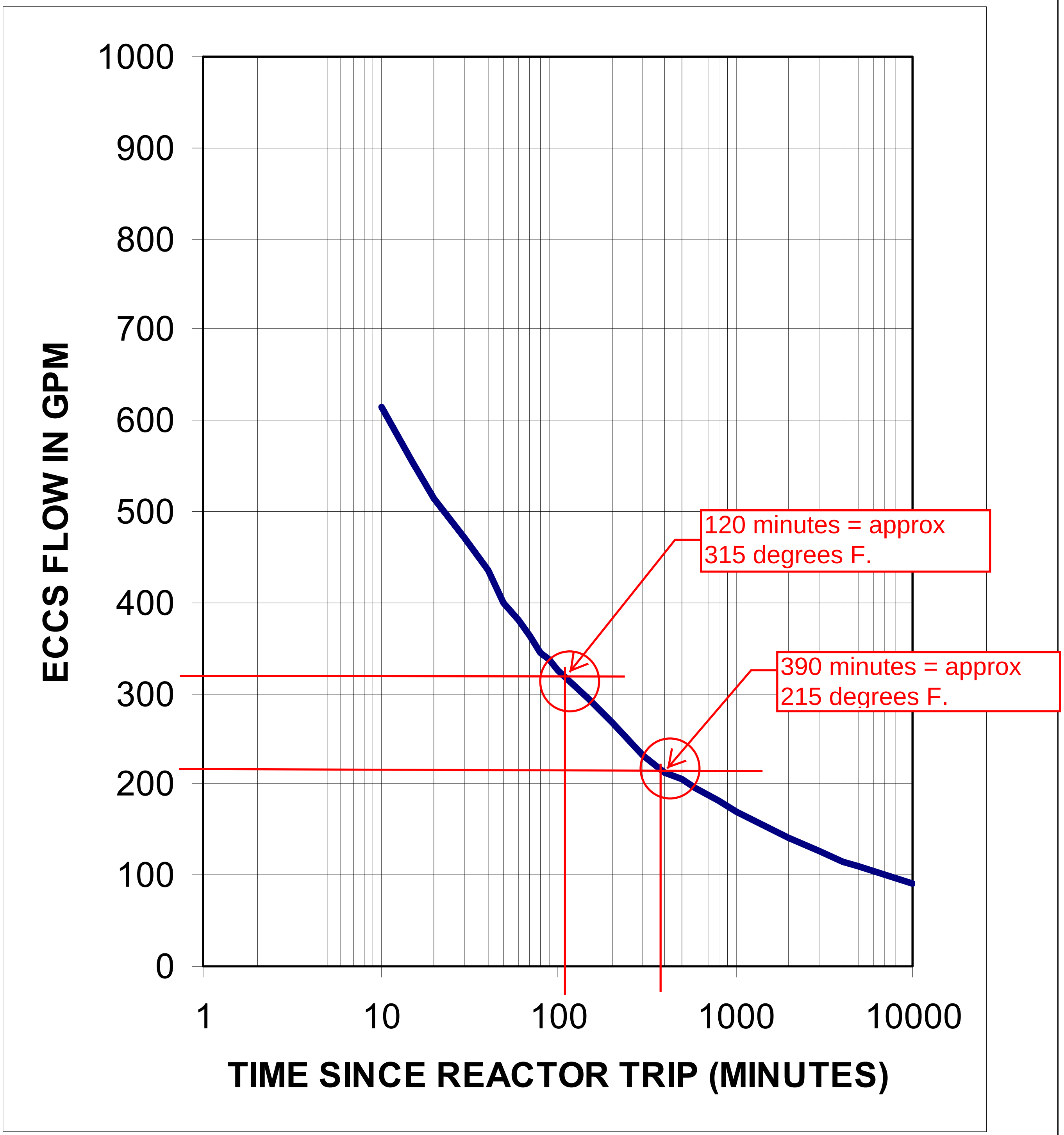
Interpolate for 120min => approximately 315gpm.

Interpolate for 390min => approximately 215gpm.

FIGURE 1

Sheet 1 of 1

MINIMUM ECCS FLOW VERSUS TIME



STEP: Check If SI Can Be Terminated

PURPOSE: To determine if conditions have been established which indicate that one train of SI flow is no longer required

BASIS:

Following the reduction to one train of SI, RCS conditions may be within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this guideline because SI flow may prevent a subsequent reduction in RCS pressure and cause considerable depletion of the RWST.

The subcooling criterion will ensure subcooled conditions and the RVLIS indication ensures the existence of an adequate vessel inventory such that core cooling is ensured. Refer to document SI TERMINATION/REINITIATION in the Generic Issues section of the Executive Volume.

If the termination criteria are not satisfied, then SI is required to ensure core cooling and should not be terminated. If RVLIS indication is adequate but RCS subcooling is not, the operator is then instructed to establish the minimum SI pump flow needed to match decay heat in order to further decrease SI pump flow and delay RWST depletion. This is done by aligning (if necessary) and operating the appropriate SI pumps (charging/SI pumps, high-head SI pumps and low-head SI pumps) such that the flow required to match decay heat is established. For most Westinghouse plants, the flow through the SI lines cannot be throttled and the exact flow rate required cannot be established. Therefore, in order to establish the minimum SI flow required in this step, the operator should stop appropriate SI pump(s) to establish flow equal to or greater than the minimum SI flow required to match decay heat. The SI flow needed to match decay heat is a function of time and is obtained from Figure 1.

Figure 1 is a generic curve with units for flowrate of gpm per MWt. Each utility must develop a plant specific curve for its plant from Figure 1, and this curve would be included in the plant specific emergency operating procedure as Figure ECA11-1. This plant specific curve can be developed by modifying Figure 1 as follows: The Y-axis values for flowrate in gpm/MWt should be multiplied by the plant specific MWt core rating to obtain flowrate values in GPM. The X-axis values for time in minutes are used without modification. A plant

specific curve is then plotted as flowrate (gpm) versus

time (minutes). Note that Figure ECA11-1 in guideline ECA-1.1 has been developed for a plant with a core rating of 3411 MWt and is included as an example.

ACTIONS:

- o Determine if RVLIS indication is greater than the full range or dynamic head range value, as applicable.
- o Determine if RCS subcooling (based on core exit TCs) is greater than (R.12)°F [(R.13)°F for adverse containment]
- o Determine minimum SI flow required from Figure ECA11-1
- o Establish minimum SI flow

INSTRUMENTATION:

- o RCS pressure indication
- o Core exit TCs temperature indication
- o RVLIS indication

CONTROL/EQUIPMENT:

- o High-head SI pump switches
- o Charging/SI pump switches
- o Low-head SI pump switches

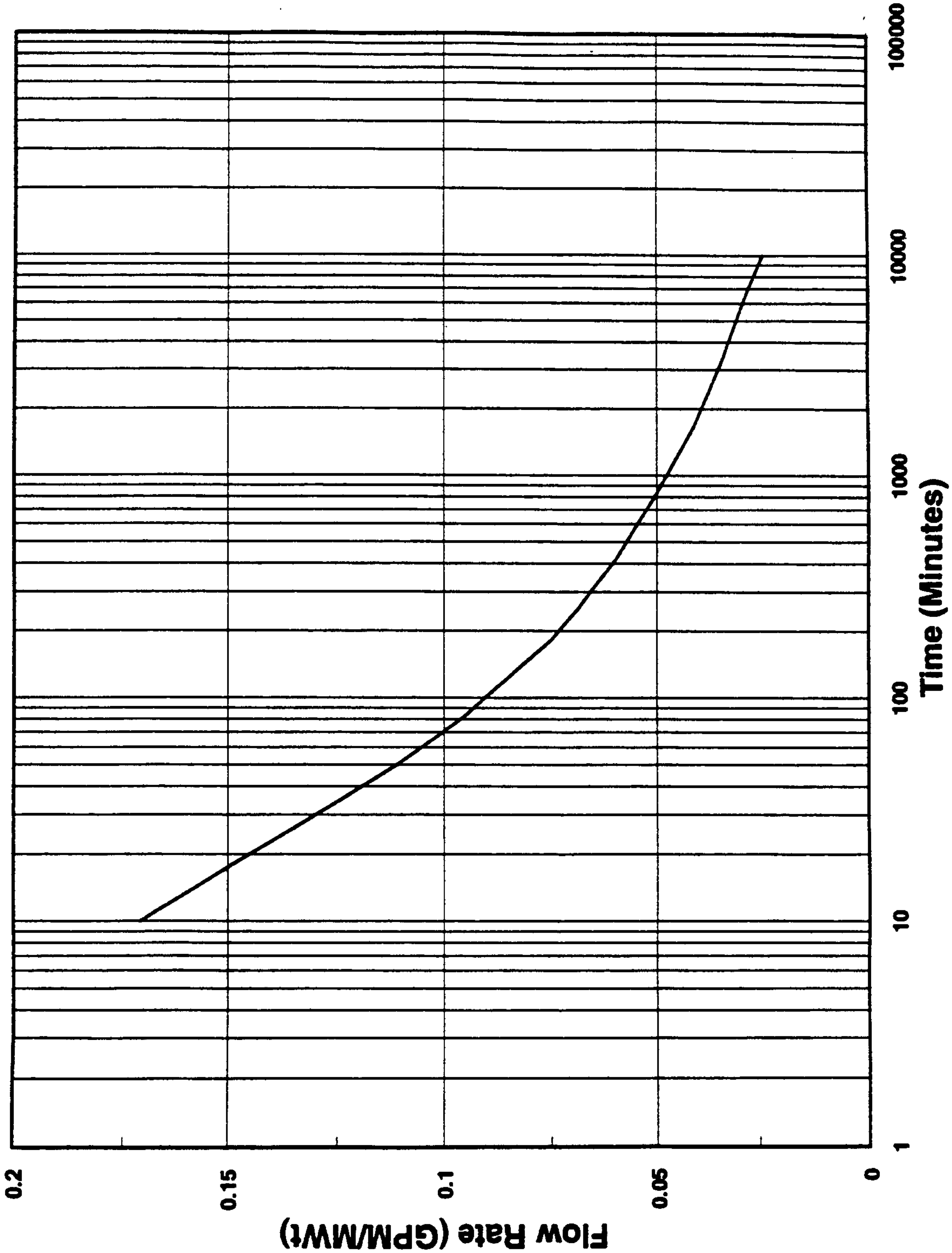
KNOWLEDGE:

- o Understanding of RVLIS function, configuration, and interpretation
- o Due to the less restrictive SI termination and reinitiation criteria provided in this guideline the operator should be especially alert for any decrease in RCS subcooling or vessel level that warrants SI reinitiation
- o This step is a continuous action step while in this guideline

PLANT-SPECIFIC INFORMATION:

- (K.02) RVLIS full range value which is the top of the core, including allowances for instrument uncertainties.
- (L.05) RVLIS dynamic range value corresponding to an average system void fraction of 25 percent with 4 RCPs running, including allowances for instrument uncertainties.
- (L.06) RVLIS dynamic range value corresponding to an average system void fraction of 25 percent with 3 RCPs running, including allowances for instrument uncertainties.
- (L.07) RVLIS dynamic range value corresponding to an average system void fraction of 25 percent with 2 RCPs running, including allowances for instrument uncertainties.
- (L.08) RVLIS dynamic range value corresponding to an average system void fraction of 25 percent with 1 RCP running, including allowances for instrument uncertainties.
- (R.12) The sum of temperature and pressure measurement system errors, including allowances for normal channel accuracies, translated into temperature using saturation tables, plus 50°F.
- (R.13) The sum of temperature and pressure measurement system errors, including allowances for normal channel accuracies and post accident transmitter errors, translated into temperature using saturation tables, plus 50°F.
- If RVLIS is not available, RCS subcooling based on core exit TCs is sufficient for terminating SI since a 50°F margin has been added to instrument uncertainties. This 50°F margin allows sufficient time for operator action to reinitiate SI before core uncovery.
- As long as the RVLIS dynamic range uncertainty for the Westinghouse RVLIS design is less than +/-6%, the uncertainty does not need to be included in the calculation of the plant-specific EOP setpoints.

Figure 1. DECAY HEAT FLOW RATE PER MWt versus TIME AFTER TRIP



1. 007A2.02 001/LOIT/RO/M/F 2.6/3.2/007A2.02/LO-PP-16301-11//HL17 NRC/

Initial conditions:

- Unit 1 is in Mode 5 with solid plant conditions.
- Pressurizer bubble is being established.

Current conditions:

- A transient results in an RCS pressure spike.
- ALB12-E02 PRZR REL TANK HI PRESS is received.
- ALB12-E03 PRZR REL TANK HI TEMP is received.

Which one of the following completes the following statement?

The RHR pump __(1)__ relief valve lifting caused the PRT high pressure condition,

and

per 13004-1, "Pressurizer Relief Tank Operation," recirculation through the RCDT heat exchanger to lower PRT temperature is expected to take approximately __(2)__ .

	__(1)__	__(2)__
A.	discharge	1 hour
B.	discharge	8 hours
C.	suction	1 hour
D.✓	suction	8 hours

K/A

007 **Pressurizer Relief / Quench Tank**

A2.02 **Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:**

Abnormal pressure in the PRT

K/A MATCH ANALYSIS

The question requires the candidate to determine the cause of the high pressure and high temperature annunciators associated with the PRT. This is backward logic from the first part of the specified K/A, which requires the candidate to predict how the malfunction will impact the PRT. This was necessary in order to write a technically valid question. The question as written requires the candidate to have knowledge of the

relationship between the malfunction and the PRT pressure. The only other possible topic that could be asked would be associated with the PRT rupture pressure. However, the rupture pressure does not lend itself to an A2 since there is no plausible distractor to pit against rupture as an impact.

The candidate is then required to choose the procedurally directed process which will correct / control the temperature and pressure increase based on the conditions depicted by the annunciators.

This question is a re-use from the HL-17 NRC exam.

EXPLANATION OF REQUIRED KNOWLEDGE

Based on the PRT annunciators, both temperature and level are elevated. Coupled with the given RCS pressure spike, the most likely source of mass addition to the PRT would be from the RHR suction relief valves, which lift at 450 psig. These valves provide cold over-pressure protection in Mode 5 and pass a significant volume of water, which will raise PRT level, pressure, and temperature. The RHR discharge relief valves are sized for thermal protection and lift at 600 psig. These valves relieve to the RHUT. With RHR pumps in service, the RHR discharge relief valves will lift first since the RHR pump creates approximately 200 psid (450psig + 200psid=650psig > 600psig). However, RCS pressure will continue to rise and the suction relief valves will open due to the small flow rate of the discharge relief valvess.

Per 17012-1 E02 and E03, PRT pressure and temperature should be restored per 13004-1. 13004-1 gives direction on how to control PRT pressure, level, and temperature. Per Limitation 2.2.1, if the PRT High Temperature Alarm is annunciated, the contents of the PRT shall be cooled. Cooling the PRT can be accomplished by two methods - spray or recirculation. Per the procedure titles and the note at the beginning of each section, cooling by spray is designed to take 1 hr and cooling by recirculation is designed to take 8 hrs. Therefore, the use of spray is fastest.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The RHR discharge valves relieve to the RHUT and therefore could not have caused the rise in PRT temperature and pressure. However, the candidate may believe the RHR discharge relief valves are located inside containment and therefore relieve to the PRT. With the added discharge pressure, the disscharge relief valves will lift before the suction relief valves and result in the described PRT conditions.

The second part is incorrect. SOP-13004 describes the 1 hour cooldown method as the spray and drain and the 8 hours method as cooldown using the RCDT heat exchanger. However, a candidate without specific knowledge of the PRT cooling methods times could find it reasonable that recirculation is a faster method since it would cool the bulk liquid temperature faster than a spray method, which tends to create thermal stratification.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. SOP-13004 describes the 1 hour cooldown method as the spray and drain and the 8 hour cooldown method as using the RCDT heat exchanger.

C. Incorrect. Plausible. The first part is correct. The RHR suction relief valves are located inside Containment and, if they lift during a RCS pressure transient, will result in the described PRT conditions.

The second part is incorrect. See the second part of choice A above

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 007A2.02
Importance Rating: 2.6 / 3.2

Technical Reference: 17012-1 Rev 21
13004-1 Rev 20
1X4DB121 Rev 42.0
1X4DB122 Rev 51.0

References provided: None

Learning Objective: LO-PP-16301-01 List the sources that input into the PRT.
LO-PP-16301-09 Describe the methods for cooling the PRT.

Question origin: MODIFIED - HL17 # 007A2.06

Cognitive Level: M/F

10 CFR Part 55 Content: 41.5 / 43.5 / 45.3 / 45.13

Comments:

You have completed the test!

1. 007A2.06 001/2/1/PRT VACUUM PROCS/H-2.6/2.8/NEW/HL-17 NRC/RO/SRO/TNT/GCW

Original Question

Initial conditions:

- Unit 1 is solid plant.
- PRZR bubble is being drawn per 12001-C, "Unit Heatup to Hot Shutdown".

Current conditions:

- A transient results in an RCS pressure spike.
- ALB12-E03 PRZR REL TANK HI TEMP illuminates

Which one of the following correctly completes the statement below?

The RHR pump ___(1)___ relief valve lifting caused the PRT High Temperature

and

per 13004-1, "Pressurizer Relief Tank Operation", the FASTEST way to cooldown the PRT is using ___(2)___.

A. (1) discharge

(2) spray from RMWST and drain to the RCDT

B. (1) discharge

(2) recirculation through the RCDT heat exchanger

C.✓ (1) suction

(2) spray from RMWST and drain to the RCDT

D. (1) suction

(2) recirculation through the RCDT heat exchanger

	(1)	(2)	(3)	(4)	(5)	(6)
A	RCP LOOP 1 LOW FLOW ALERT		RC LOOP ΔT/AUCT ΔT HI-LO DEV	RC LOOP TAVG/AUCT TAVG HI-LO DEV	TAVG/TREF DEVIATION	OVERTEMP ΔT ALERT
B	RCP LOOP 2 LOW FLOW ALERT			AUCT TAVG HIGH	TAVG LO-LO ALERT	OVERPOWER ΔT ALERT
C	RCP LOOP 3 LOW FLOW ALERT			A COLD OP LOW AUCT RCS TEMP	B COLD OP LOW AUCT RCS TEMP	TERR (TAVG - TREF) LO
D	RCP LOOP 4 LOW FLOW ALERT		PRZR PRESS LO PORV BLOCK	A RCS PRESS APPROACHES COLD OP LIMIT	B RCS PRESS APPROACHES COLD OP LIMIT	RV VENT HI TEMP
E	PRZR RELIEF DISCH HI TEMP	PRZR REL TANK HI PRESS	PRZR REL TANK HI TEMP	PV-0455A OPEN SIGNAL		A COLD OP ACTU VLV HV-8000A NOT FULL OPEN
F	PRZR SAFETY RELIEF DISCH HI TEMP	PRZR REL TANK HI/LO LEVEL	RV FLG LKOF HI TEMP	PV-0456A OPEN SIGNAL		B COLD OP ACTU VLV HV-8000B NOT FULL OPEN

WINDOW E02

ORIGIN	**SETPOINT**	PRZR REL TANK HI PRESS
1-PT-0469	8 psig	

1.0 PROBABLE CAUSE

1. One or more of the following valves has lifted or is leaking to the Pressurizer Relief Tank:

    a. Pressurizer Safety Valves,

    b. Pressurizer (PRZR) Power Operated Relief Valves (PORV)s,

    c. Chemical and Volume Control System (CVCS) Letdown Relief Valve 1-PSV-8117,

    d. CVCS Seal Return Relief Valve 1-PSV-8121,

    e. Residual Heat Removal (RHR) Relief Valves 1-PSV-8708A and B during shutdown conditions.

2. Nitrogen Regulator malfunction.

3. Safety grade letdown in use and aligned to the Pressurizer Relief Tank.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

WINDOW E02
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

If PRT pressure increases the PRT rupture disk will fail at 86 to 100 psig, opening the PRT to containment. ☐

1. **Determine** actual Pressurizer Relief Tank pressure using 1-PI-0469 on the QMCB.

2. **Monitor** Pressurizer Relief Tank temperature, level, and pressure.

3. **Check** tailpipe temperatures for the Pressurizer Safety Valves, Power Operated Relief Valves, and CVCS Letdown Relief Valve.

4. IF a PRZR PORV OR Safety Valve has actuated, **check** valve closure when pressure is lowered in the Reactor Coolant System.

5. IF a nitrogen supply malfunction has occurred, **isolate** the supply by shutting valves 1-HV-8033 and 1-HV-8047.

6. IF a Pressurizer Safety Valve is open OR fails to close following an actuation, **Go To** 18004-C, "Reactor Coolant System Leakage."

7. IF a PRZR PORV 455A/456A is open OR fails to close following an actuation:

   a. **Place** the Control Switch for the affected valve to the closed position,

   b. IF the affected valve will NOT close, **close** the associated Block Valve,

   c. **Refer** to Technical Specification LCO 3.4.11.

8. IF the pressure rise is due to the CVCS Letdown Relief Valve being open **isolate** letdown, and **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

WINDOW E02
(Continued)

9. IF the pressure rise is due to a failed RHR Relief Valve, **isolate** the affected Train of RHR and **initiate** 18019-C, "Loss Of Residual Heat Removal."

10. IF the pressure rise is due to a failed Seal Return Relief Valve, attempt to **isolate** the leak.

11. IF pressure rise is due to a hard bubble, i.e., no temperature or level change, **notify** Chemistry and **place** Pressurizer Steam Space Sample in service and **control** RCS pressure using 12004-C.

12. **Restore** pressure in the Pressurizer Relief Tank to normal per 13004-1, "Pressurizer Relief Tank Operation."

13. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, PLS

WINDOW E03

ORIGIN	SETPOINT	PRZR REL TANK HI TEMP
1-TE-0468	115°F	

1.0 **PROBABLE CAUSE**

1. One or more of the following valves has lifted or is leaking to the Pressurizer Relief Tank:

   a. Pressurizer Safety Valve,

   b. Pressurizer (PRZR) Power Operated Relief Valves (PORV)s,

   c. Chemical and Volume Control System (CVCS) Letdown Relief Valve 1-PSV-8117,

   d. CVCS Seal Return Relief Valve 1-PSV-8121,

   e. Residual Heat Removal (RHR) Relief Valves 1-PSV-8708A and B during shutdown conditions.

2. Safety grade letdown in use and aligned to the Pressurizer Relief Tank.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Determine** the actual temperature of the Pressurizer Relief Tank using 1-TI-0468 on the QMCB.

2. **Check** tailpipe temperatures for the Pressurizer Safety Valves, PRZR PORVs, and CVCS Letdown Relief Valve.

WINDOW E03
(Continued)

3. IF a Pressurizer Safety OR PRZR PORV Valve has actuated, **check** valve closes when system pressure is reduced.

4. IF a Pressurizer Safety Valve is open OR fails to close following an actuation, **initiate** 18004-C, "Reactor Coolant System Leakage."

5. IF a PRZR PORV 455A/456A is open OR fails to close following an actuation:

   a. **Place** the Control Switch for the affected valve to the closed position,

   b. IF the affected valve will NOT close, **close** the associated Block Valve,

   c. **Refer** to Technical Specification LCO 3.4.11.

6. IF the temperature rise is due to the CVCS Letdown Relief Valve being open, **isolate** letdown and **initiate** 18007-C, "Chemical And Volume Control System Malfunction."

7. IF the temperature rise is due to a failed RHR Relief Valve, **isolate** the affected Train of RHR and **initiate** 18019-C, "Loss Of Residual Heat Removal."

8. IF the temperature rise is due to a failed Seal Return Relief Valve, attempt to **isolate** the leak.

9. **Restore** the Pressurizer Relief Tank temperature to normal per 13004-1, "Pressurizer Relief Tank Operation."

10. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, PLS

1.0 PURPOSE

This procedure provides the necessary instructions for operation of the Pressure Relief Tank (PRT) and supporting equipment. Procedure instructions include the following:

4.1.2 Placing The PRT In Service

4.2.1 Pressure Control Of The PRT

4.2.2 Level Control Of The PRT

4.4.1 Purging The PRT To The WGS

4.4.2 Purging The PRT To Containment HVAC

4.4.3 PRT Cooldown Using Spray And Drain (One Hour Cooldown)

4.4.4 PRT Cooldown Using RCDT Heat Exchanger (Eight Hour Cooldown)

4.4.5 Draining The PRT

4.4.6 Venting PRT For PRZ Code Safeties OR Manway removal

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 Equipment that has been exposed to air must be purged before it is connected to the Vent Header to prevent explosive mixtures of hydrogen and oxygen. ________

2.1.2 A nitrogen gas blanket should be maintained in the PRT to exclude air and prevent the formation of an explosive hydrogen and oxygen mixture. ________

2.1.3 Before opening the PRT to atmosphere, verify that the gas space hydrogen content is less than 4% and the oxygen content is less than 5%. ________

2.2 LIMITATIONS

2.2.1 If the PRT High Temperature Alarm (115°F) is annunciated, the contents of the PRT shall be cooled. ________

2.2.2 The PRT rupture disk will fail at 86 to 100 PSIG. ________

3.0 PREREQUISITES AND INITIAL CONDITIONS

3.1 The Gaseous Waste Processing System is available to provide processing of gases from the PRT. ________

3.2 The Auxiliary Gas System-Nitrogen or nitrogen from the Waste Gas Decay Shutdown Tank is available to provide a nitrogen blanket for the PRT. ________

3.3 Reactor Make-Up Water is available to provide a cooling spray for the PRT. ________

3.4 ACCW is available if cooling the PRT with the RCDT Heat Exchanger. ________

INITIALS

4.4.3 PRT Cooldown Using Spray And Drain (One Hour Cooldown)

NOTE

Two methods for cooling the PRT exist. Cooling the PRT by spray and drain is designed to cool the PRT in 1 hour. This method uses makeup water and drains to the Waste Processing System. Cooling the PRT by recirculation through the RCDT Hx is designed to cool the PRT in 8 hours. This method minimizes makeup water use and waste processing of liquid. The time required to cool the PRT and water usage should be considered before deciding which method to use.

4.4.3.1 **Establish** communications between the Liquid Waste Processing System Panel (PLPP) and the Control Room. ________

4.4.3.2 **Verify** the PRT pressure less than or equal to 50 psig as indicated by PRESSURIZER RELIEF TANK 1-PI-0469 to prevent RCDT System over pressurization. ________

4.4.3.3 **Verify** open WPSL RCDT PUMPS DISCH TO RECYC EVAP 1-1901-U6-327. (1AB-RA27) ________

4.4.3.4 **Realign** RCDT Pump Suction to the PRT and **initiate** spray as follows:

a. **Stop** the running REACTOR COOLANT DRAIN TANK PUMP

#1 1HS-1003A (PLPP) ________

#2 1HS-1003B (PLPP) ________

CAUTION

The RCDT level should be monitored to prevent tank flooding.

b. **Place** REACTOR COOLANT DRAIN TANK LEVEL 1-LC-1003 in MANUAL and **open** the valve (PLPP). ________

c. **Close** REACTOR COOLANT DRAIN TANK RECIRCULATION VALVE 1-HV-7144 (PLPP). ________

INITIALS

4.4.4 PRT Cooldown Using RCDT Heat Exchanger (Eight Hour Cooldown)

NOTE

Two methods for cooling the PRT exist. Cooling the PRT by spray and drain is designed to cool the PRT in 1 hour. This method uses makeup water and drains to the Waste Processing System. Cooling the PRT by recirculation through the RCDT Hx is designed to cool the PRT in 8 hours. This method minimizes makeup water use and waste processing of liquid. The time required to cool the PRT and water usage should be considered before deciding which method to use. □

4.4.4.1 **Establish** communications between the Liquid Waste Processing System Panel (PLPP) and the Control Room. ________

4.4.4.2 **Verify** PRESSURIZER RELIEF TANK 1-PI-0469 indicates less than OR equal to 50 psig to prevent RCDT System over pressurization. (QMCB) ________

4.4.4.3 **Realign** the RCDT System to recirc the PRT as follows:

a. **Stop** the running REACTOR COOLANT DRAIN TANK PUMP

#1 1HS-1003A (PLPP) ________

#2 1HS-1003B (PLPP) ________

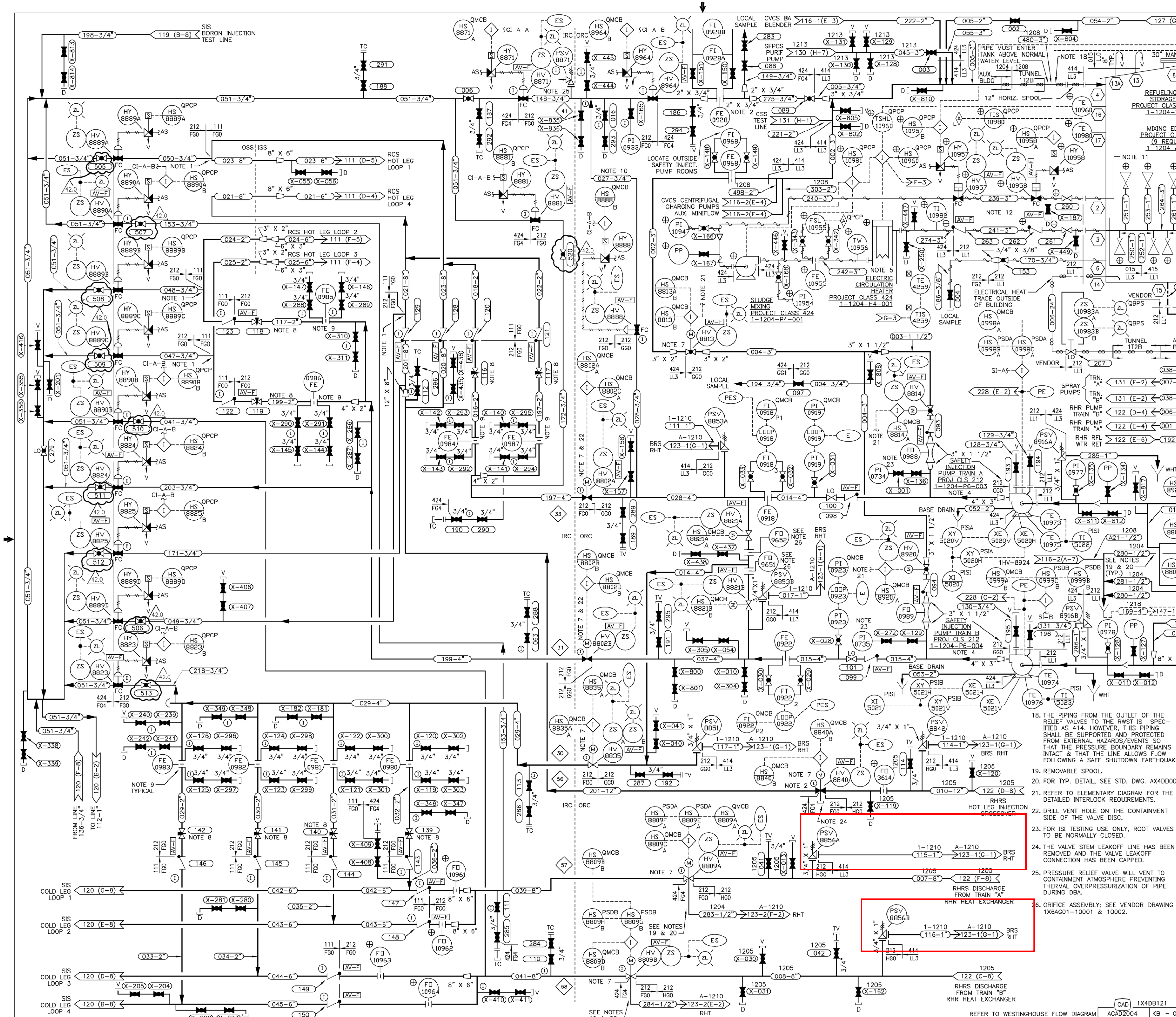
CAUTION

The RCDT level should be monitored to prevent tank flooding. □

b. **Close** REACTOR COOLANT DRAIN TANK PUMP SUCTION VALVE 1-HV-7127 (PLPP). ________

c. **Close** REACTOR COOLANT DRAIN TANK RECIRCULATION VALVE 1-HV-7144 (PLPP). ________

d. **Open** REACTOR COOLANT DRAIN TANK PRESSURE RELIEF TANK VALVE 1-HV-7141 (PLPP). ________



REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E71 SHEET 3 OF 3.

NOTES:

1. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
2. FOR DRAIN HEADER INFORMATION SEE DWG 1X4DB143 & 144-1.
3. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT (1) EQUIPMENT MARKED ⊕; (2) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
4. FOR COOLING CONNECTIONS SEE DWG 1X4DB134.
5. ELECTRIC CIRCULATION HEATER TO MAINTAIN MINIMUM 50° F WATER TEMP IN RWST.
6. LOCATE TAPS AT SAME ELEVATION. PROVIDE MAXIMUM PHYSICAL SEPARATION BETWEEN EACH TAP LOCATION.
7. PROVIDE FOR POWER LOCKOUT AND RESTORATION FROM CONTROL ROOM.
8. ADJUST VALVES IN FIELD TO LIMIT PUMP RUNOUT AND EQUALIZE FLOWS, THEN LOCK IN POSITION.
9. FLOW METERING ORIFICE TO VERIFY FLOW DURING PRE-OPERATIONAL TESTING.
10. LOCATE DRAIN AT LOW POINT BETWEEN VALVES HV8871, HV8888, AND 089.
11. THIS SUCTION PIPE INLET NOZZLE SHALL BE LOCATED APPROXIMATELY 11 FEET ABOVE THE RWST BOTTOM.
12. THESE CLASS 212 VALVES AND THEIR CONTROLS SHALL BE PROTECTED BY A MISSILE SHIELD.
13. ALL EDUCTORS SHALL BE LOCATED AT THE SAME ELEVATION, BETWEEN 12" AND 18" ABOVE THE RWST BOTTOM.
14. PORTION OF THE SYSTEM SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
15. LOCATE VALVES HV8806, 090, 135 AND 136 AS CLOSE AS POSSIBLE.
16. ALL SAFETY-RELATED PROCESS, INSTRUMENT AND SAMPLING LINES THAT ARE PHYSICALLY ROUTED IN AN AREA EXPOSED TO EXTREMELY COLD WEATHER SHALL BE PROVIDED WITH ADEQUATE HEAT TRACING.
17. ⬡ INDICATES TANK NOZZLE CONNECTION NUMBER PER PLANT DESIGN DWG 1X4DL57F02 REV. 4.
18. THE PIPING FROM THE OUTLET OF THE RELIEF VALVES TO THE RWST IS SPECIFIED AS 414. HOWEVER, THIS PIPING SHALL BE SUPPORTED AND PROTECTED FROM EXTERNAL HAZARDS/EVENTS SO THAT THE PRESSURE BOUNDARY REMAINS INTACT & THAT THE LINE ALLOWS FLOW FOLLOWING A SAFE SHUTDOWN EARTHQUAKE.
19. REMOVABLE SPOOL.
20. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
21. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK REQUIREMENTS.
22. DRILL VENT HOLE ON THE CONTAINMENT SIDE OF THE VALVE DISC.
23. FOR ISI TESTING USE ONLY, ROOT VALVES TO BE NORMALLY CLOSED.
24. THE VALVE STEM LEAKOFF LINE HAS BEEN REMOVED AND THE VALVE LEAKOFF CONNECTION HAS BEEN CAPPED.
25. PRESSURE RELIEF VALVE WILL VENT TO CONTAINMENT ATMOSPHERE PREVENTING THERMAL OVERPRESSURIZATION OF PIPE DURING DBA.
26. ORIFICE ASSEMBLY; SEE VENDOR DRAWING 1X6AG01-10001 & 10002.

Q

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
SAFETY INJECTION SYSTEM
SYSTEM NO. 1204

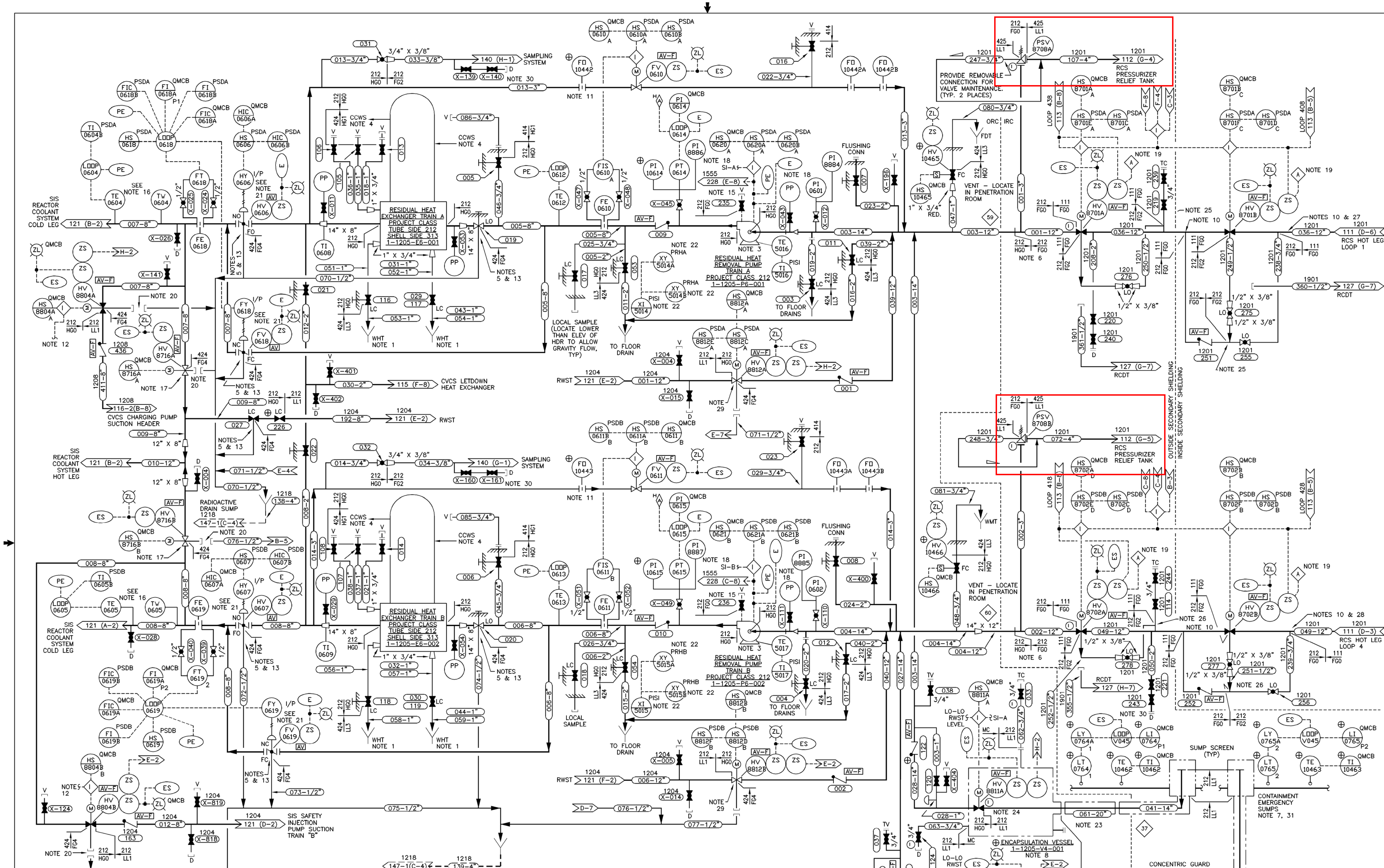
CAD 1X4DB121 | ACAD2004 | KB - 02

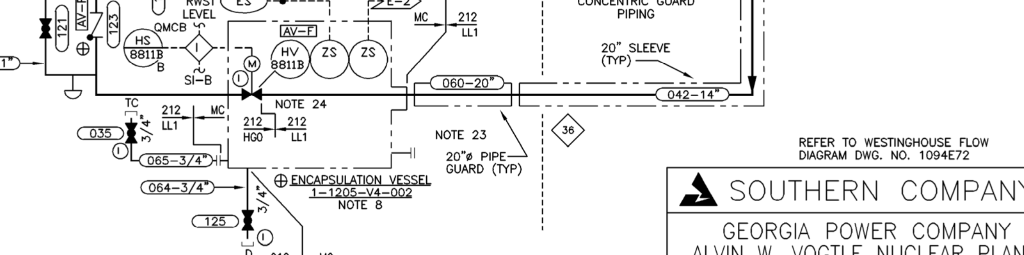
NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
42.0	REVISED PER ABN-1071073501M004, VER. 1.0	10/20/09	KB	CFC	JMR	X

SCALE:	DRAWING NO.	VER.
NONE	1X4DB121	42.0
JOB NO.10604		

SIZE E 34X44

DRAWING CATEGORY: CRITICAL





NOTES:

1. FOR DRAIN HEADER INFORMATION SEE DWGS 1X4DB143, 144-1, 1X4DB145-5 AND 1X4DB183.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT: (1) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER, AND (2) EQUIPMENT MARKED ⊕.
3. FOR PUMP AND MOTOR COOLING WATER CONNECTIONS, SEE DWGS 1X4DB134 & 1X4DB137.
4. FOR HEAT EXCHANGER SHELL SIDE CONNECTIONS SEE DWG 1X4DB137.
5. PROVIDE 12" REMOVABLE SPOOL.
6. MAKE PIPING MATERIAL CLASS CHANGE AT FIRST WELD OF PENETRATION SLEEVE INSIDE THE CONTAINMENT.
7. INSTALL 150 #R.F. FLANGE ON INLET OF EACH 14"ø PIPE FOR PERIODIC TESTING. PROVIDE ONE MATING BLIND FLANGE TO BE DRILLED AND TAPPED FOR TEST CONNECTION. MATING FLANGE MUST BE REMOVED FOR NORMAL OPERATION.
8. VESSEL IS CONSTRUCTED TO SEISMIC CATEGORY 1 AND ASME III CLASS MC.
9. PORTIONS OF THIS SYSTEM ARE SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON DWG CX4DG001. 51.0
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. FOR TYPICAL DETAIL SEE STD. DWG AX4DD000.
14. DELETED.
15. HIGH POINT VENT VALVE (BECHTEL-FURNISHED) FOR SEAL PIPING - SEE 1X6AF02-25.
16. TE IS A SURFACE TYPE CLAMP ON RTD, TW IS INSTALLED BUT NOT USED.
17. VENT HOLE IS DRILLED ON RESPECTIVE RHR PUMP SIDE OF DISC.
18. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.
19. ALARM ON OPEN HOT LEG VALVE AND HIGH RCS PRESSURE.
20. DELETED.
21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.
23. 4" VIEW PORT TO BE USED TO VERIFY MOTOR OPERATED VALVE STROKE TEST WITHOUT REMOVING THE VESSEL HEAD.
24. VENT HOLE IS DRILLED ON CONTAINMENT SIDE OF DISC.
25. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-249-01.
26. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-251-01.
27. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-036-01.
28. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-049-02.
29. A SECTION OF THE LEAKOFF LINE HAS BEEN REMOVED AND BOTH ENDS CAPPED.
30. THESE ARE KEROTEST VALVES AND MAY NOT ALLOW REVERSE FLOW.
31. SUMP SCREEN TAG NO'S. 1-1205-G2-001 (TRAIN "A") AND 1-1205-G2-002 (TRAIN "B")

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. NO. 1094E72

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
51.0	REVISED PER ABN-V02177, VERSION 1.0	05/19/10	LPM	LCF	PML	X

CAD 1X4DB122
ACAD2007 | LPM - 07

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
RESIDUAL HEAT REMOVAL
SYSTEM NO. 1205

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB122	51.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 007EA2.06 001/LOIT AND LOCT/RO/C/A 4.3/4.5/007EA2.06/LO-TA-37002/27009///

Initial condition:

- Unit 1 is operating at 55% reactor power with a plant startup in progress.

Current conditions:

- Both Unit 1 RATs de-energize.
- Both DGs start and re-energize their respective bus.
- No other switchyard components are affected.

Which one of the following completes the following statement?

With no operator action, **one minute after** the RATs de-energize, the Reactor Trip Breakers will be __(1)__,

and

DRPI __(2)__ be available to check control rod positions.

	__(1)__	__(2)__
A.	open	will
B.	open	will NOT
C.✓	closed	will
D.	closed	will NOT

K/A

007 **Reactor Trip - Stabilization - Recovery**

EA2.06 **Ability to determine or interpret the following as they apply to a reactor trip:**

Occurrence of a reactor trip.

K/A MATCH ANALYSIS

The question presents the candidate with a loss of both RATs and then requires the candidate to determine if a reactor trip occurs as a result of the LOSP, and whether DRPI will be energized after the busses are reenergized by the EDGs.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to recognize that above 50% power, the non-1E 13.8 KV and 4160 VAC buses will remain energized from the UATs. The 1E 4160 VAC buses will de-energize and be re-energized by the Emergency Diesel Generators (EDGs). As such, no reactor trip signals will be generated directly or indirectly. Additionally, DRPI is normally powered from 'B' train 1E power (1BA03->1BBC->1NYC2).

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. As described in the Explanation of Required Knowledge above, all buses are energized after the loss of the RATs and no reactor trip signals are generated. However, below 50% power the non-1E buses are aligned to the RATs. A candidate not familiar with when the non-1E buses are transferred could conclude that the non-1E buses de-energized. In this case, several reactor trip signals would be energized.

The second part is correct. 1BA03 was re-energized by the DG and the sequence re-energized 1BBC, which re-energized DRPI.

B. Incorrect. Plausible. The first part is incorrect. See part one of Choice A above.

The second part is incorrect. As described in the Explanation of Required Knowledge above, DRPI is ultimately powered from 1BA03, which will be re-energized by its EDG. However, DRPI is powered from a non-1E panel 1NYC2. If the candidate is not familiar with the one-line distribution for DRPI, it could be reasonable for a candidate, who concluded that non-1E buses de-energized in part one of this distractor, to also conclude that DRPI, powered from a non-1E 120VAC panel, would also be de-energized.

C. Correct. The first part is correct. As described in the Explanation of Required Knowledge above, all buses are energized after the loss of the RATs and no reactor trip signals are generated.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. As described in the Explanation of Required Knowledge above, DRPI is ultimately powered from 1BA03, which will be re-energized by its EDG. It is reasonable for a candidate without specific knowledge of DRPI power supplies to recongnize DRPI is de-energized when 1BA03 losses power during scenarios and conclude DRPI will not be automatically restored on a loss of power, because DRPI is non-safety related and there are no steps in the EOPs to ensure it is energized.

Level:	RO	
Tier # / Group #	T1 / G1	
K/A#	007EA2.06	
Importance Rating:	4.3 / 4.5	
Technical Reference:	13432-1 Rev 49	
References provided:	None	
Learning Objective:	LO-PP-28103-03	List all reactor trip set points, coincidences, permissives, and blocks.
	LO-PP-28103-05	List all ESF actuation signals with applicable set points, coincidences, permissives, blocks, and discuss the systems response to each ESF actuation signal.
	LO-PP-27201-03	State the power supplies for the DRPI System.
	LO-TA-37002	Respond to a Reactor Trip Without Safety Injection per 19000-C and 19001-C
	LO-TA-27009	Swap DRPI Power Supplies using 13432-1/2
Question origin:	BANK - FNP EXAM BANK Question # 007EA2.06 007	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.7 / 45.5 / 45.6	
Comments:		

You have completed the test!

A. LOSS OF POWER WITH DG FAILING TO TIE TO BUS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

A14. Verify DRPI - ENERGIZED.

A14. Perform the following:

a. Swap DRPI power supply using 13432, 120V AC NON 1E INSTRUMENT DISTRIBUTION SYSTEM.

UNIT	LOCATION	NORMAL SUPPLY	ALTERNATE SUPPLY
1	1NYC2 (CB-B66)	1BBC-20	1ABC-20
2	2NYC2 (CB-B12)	2BBC-20	2ABC-20

b. IF DRPI can NOT be energized, THEN refer to the following technical specifications as applicable:

TS 3.0.3
TS 3.1.7
TR 13.1.8
TR 13.1.9

*A15. **Check DC bus loads:**

a. Verify 125V DC battery loads - LESS THAN THE FOLLOWING LIMITS:

- AD1B 300 AMPS
- BD1B 300 AMPS
- CD1B 100 AMPS
- DD1B 80 AMPS

a. Evaluate selective load stripping using ATTACHMENT B, DC LOADS TO EVALUATE FOR LOAD STRIPPING DURING LOSS OF 1E BUS.

Step 15 continued on next page

INITIALS

4.2.3 Transferring Regulated Instrument Distribution Panel 1NYC2, 1NYJ, 1NYR, 1NYS or 1NYRS to Alternate Source

NOTES

- The normal regulated transformer is the preferred power source for the instrument distribution panels. ☐
- Any TS LCO or TRM actions entered as a result of panels being de-energized should be evaluated for exit after panel is re-energized and loads verified operable. ☐

CAUTION

When power is transferred for 1NYC2, DPM 1RX-12444 may be damaged if not removed from service by Chemistry. ☐

4.2.3.1 To transfer 1NYC2 to the alternate regulated source 1ABC, **perform** the following:

a. **Request** Chemistry remove power at DPM 1RX-12444 to support transfer. ________

Critical

b. **Close** the Alternate Regulated Transformer Supply Breaker 1ABC-20. (CB-B76) ________

____
CV

INITIALS

NOTE

Steps 4.2.3.1.c and 4.2.3.1.d should be performed as quickly as possible to minimize the time the panel is de-energized. ☐

c. **Open** the normal Instrument Distribution Panel Breaker 1NYC2-02. (CB-B66) ________

d. **Close** the alternate Instrument Distribution Panel Breaker 1NYC2-01. (CB-B66) ________

e. **Verify** proper operation of 1NYC2 by observing approximately 120V AC on the Instrument Distribution Panel Voltmeter. ________

Critical

f. WHEN 1NYC2 is transferred to the Alternate Source, **open** the Normal Regulated Supply Breaker 1BBC-20. (CB-61) ________

________ CV

g. **Request** Chemistry restore power to DPM 1RX-12444 following energization. ________

INITIALS

4.2.4 Transferring Regulated Instrument Distribution Panel 1NYC2, 1NYJ, 1NYR, 1NYS, or 1NYRS to Normal Source

NOTES

- The normal regulated transformer is the preferred power source for the instrument distribution panels. ☐
- Any TS LCO or TRM actions entered as a result of panels being de-energized should be evaluated for exit after panel is re-energized and loads verified operable. ☐

CAUTION

When power is transferred for 1NYC2, DPM 1RX-12444 may be damaged if not removed from service by Chemistry. ☐

4.2.4.1 To transfer 1NYC2 to the normal regulated source 1BBC, **perform** the following:

a. **Request** Chemistry remove power at DPM 1RX-12444 to support transfer. ________

<u>**Critical**</u>

b. **Close** the Normal Regulated Transformer Supply Breaker 1BBC-20. ________

________ CV

NOTE

Steps 4.2.4.1.c and 4.2.4.1.d should be performed as quickly as possible to minimize the time the panel is de-energized. ☐

c. **Open** the alternate Instrument Distribution Panel Breaker 1NYC2-01. ________

INITIALS

d. **Close** the normal Instrument Distribution Panel Breaker 1NYC2-02. ________

e. **Verify** proper operation of 1NYC2 by observing approximately 120V AC on the Instrument Distribution Panel Voltmeter. ________

Critical

f. WHEN 1NYC2 is transferred to the normal Source, **open** the alternate Regulated Supply Breaker 1ABC-20. ________

____
CV

g. **Request** Chemistry restore power at DPM 1RX-12444 following energization. ________

1. 008A1.03 001/LOIT/RO/C/A 2.7/2.9/008A1.03/LO-TA-60026///

Initial conditions:

- Unit 1 is at 100% reactor power.
- CCW Train 'A' is in service.
- Multiple CCW Train 'A' low pressure and low flow alarms annunciate, then clear.
- No other alarms were received.

Current conditions:

- CCW Train 'A' pressure and flow are stable.
- 18020-C, "Loss of Component Cooling Water," is entered.

Which one of the following completes the following statement?

A CCW Train 'A' pump has experienced a __(1)__,

and

the standby CCW Train 'A' pump __(2)__ auto start in response to CCW low header pressure.

	__(1)__	__(2)__
A.	locked rotor	did
B.	locked rotor	did NOT
C.✓	sheared shaft	did
D.	sheared shaft	did NOT

K/A

008 **Component Cooling Water**

A1.03 **Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including:**

CCW pressure

K/A MATCH ANALYSIS

The KA addresses the relationship between the CCW System and the operator's ability to monitor the system, specifically including pressure, and make decisions on the proper response and actions. The question requires the candidate to evaluate various indications and observed system responses, and determine whether the system is responding correctly to the malfunction or if operator intervention is required.

Note: Unlike the typcial CCW system on most Westinghouse plants, the CCW system is split into two systems on Vogtle 1&2 - CCW and ACCW. CCW only contains SFP Cooling, RHR Hx, and RHR seal cooler. As such, both CCW and ACCW are very simple. The only active valves in the CCW system are the pump discharge check valves. The only control circuits are associated with pump starts and trips. ACCW, which only supports non-essential loads, additionally has active valves around the RCP thermal barriers but these are already addressed on a different question in this exam. Therefore, this question focus is the best overall fit for the K/A.

EXPLANATION OF REQUIRED KNOWLEDGE

If a shaft shear occurs on one of the two operating CCW pumps, CCW system flow and pressure decrease and bring in multiple system alarms. With at least one CCW handswitch in AUTO, the low pressure auto start is enabled. The standby CCW pump will start and restore system pressure and flow to normal, clearing all the annunciators. All three CCW pump motors will be running. The pump with the sheared shaft will not trip.

In contrast, a CCW pump with a locked rotor will produce the same sequence of events, except the handswitch on the pump with the locked rotor will have amber and green lights lit. Additionally, a QEAB annunciator will alarm when the associated 4160V breaker trips. In this case, the standby pump will start when the 52 contact changes state and not on low header pressure. The 52 contact change is instantaneous, whereas the low header pressure start requires system pressure to decay over a period of time.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The candidate could determine the low pressure alarms are consistent with the trip of a running pump and the system response is to auto start the standby pump, rather than the pressure decay causing the pump to start on low pressure. The candidate may not consider the absence of electrical alarms in this condition when making their decision.

The second part is correct. Per the Explanation of Required Knowledge above, the standy CCW pump would start on a low header pressure signal with a shaft shear on the affected pump.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per the Explanation of Required Knowledge above, the standy CCW pump will start on a low header pressure signal with the affected pump shaft sheared. However, if the candidate determined the start signal was a result of the trip of the running pump, the standby pump would have started directly from the 52 contact prior to a low pressure condition existing.

C. Correct. The first part is correct. The annunciators came in and then cleared, informing the candidate that actions have taken place to correct the malfunction. In addition, the candidate should recognize the absence of QEAB alarms and the handswitch light indication as symptoms of a sheared shaft as opposed to a locked rotor.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 008A1.03
Importance Rating: 2.7 / 2.9

Technical Reference: 1X3D-BD-L01A Rev 12.0
ARP 17002-1 Rev 24.1
ARP 17036-1 Rev 21.0

References provided: None

Learning Objective: LO-PP-10101-04 From memory, describe the expected system response and operator corrective actions for each of the following:

a. SI
b. LOSP
c. SI with LOSP
d. Surge Tank Low level
e. Low header pressure
f. Pump shaft shear/locked rotor
g. Three pumps running

LO-TA-60026 Respond to a Loss of CCW per 18020-C

Question origin: MODIFIED - HL15 Audit Question # 038EA1.25 001

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 45.5

Comments:

You have completed the test!

1. 038EA1.25 001/1/1/SGTR - CCW AMPS/C/A - 2.6/NEW/RO/HL-14 AUDIT/TNT / RLM

Initial Unit 1 conditions:

- 19031-C, "ES-3.1 Post-SGTR Cooldown Using Backfill" is in use.
- RHR pump A has been placed in service in shutdown cooling mode.

Current conditions:

- Several CCW Train A low flow and pressure alarms annunciate, then clear.
- All 3 CCW Train A pumps red lights are lit.
- CCW Train A system has normal flows and pressures.

A CCW pump __(1)__ has occurred.

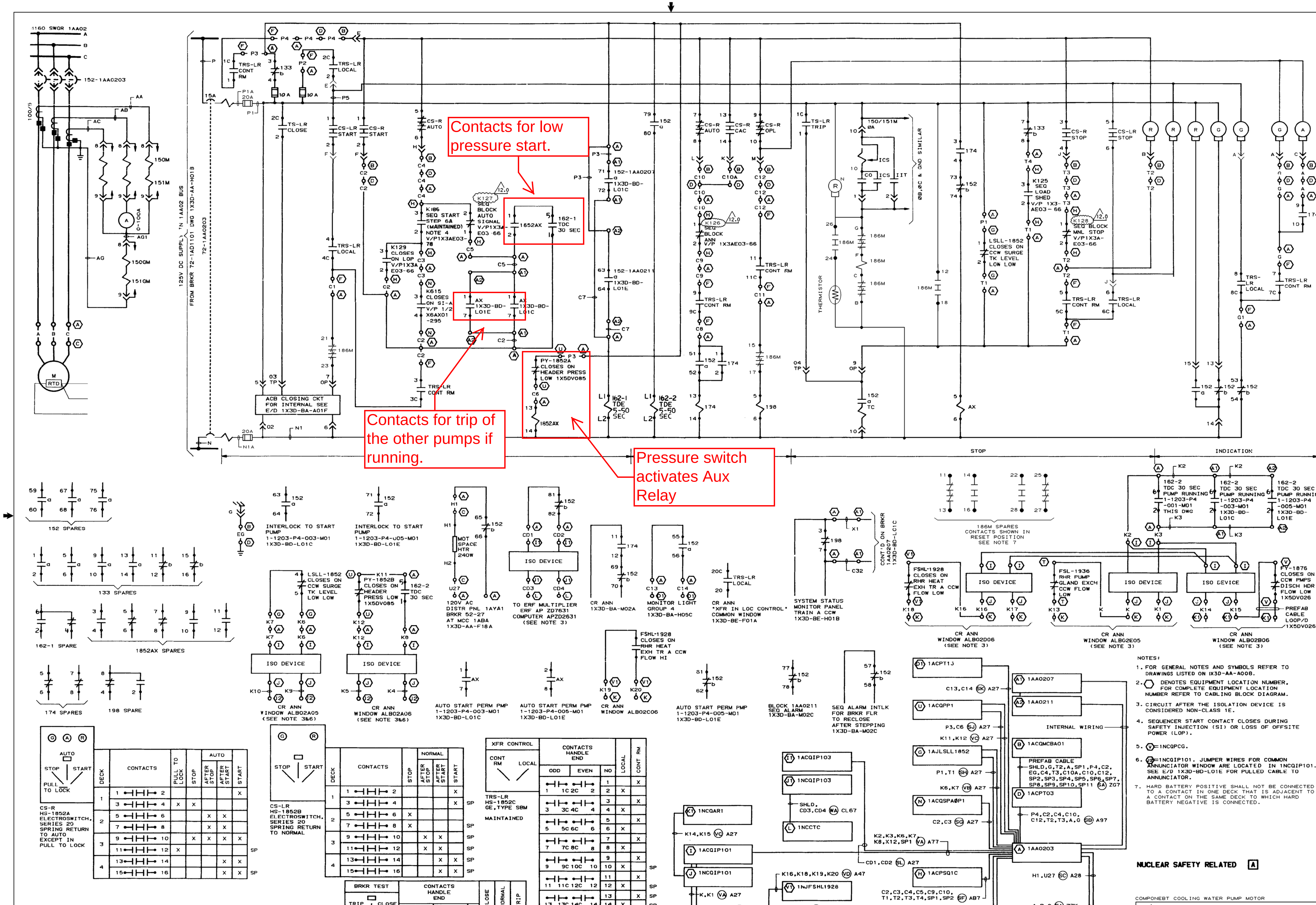
You must monitor pump amps __(2)__ in order to determine which pump to stop.

	(1)	(2)
A.	locked rotor	on the QEAB
B.	shaft shear	on the QEAB
C.	locked rotor	locally at 1AA02
D.✓	shaft shear	locally at 1AA02

	(1)	(2)	(3)	(4)	(5)	(6)
A	**NSCW TRAIN A F-1 HI VIB**	**NSCW TRAIN A F-2 HI VIB**	**NSCW TRAIN A F-3 HI VIB**	**NSCW TRAIN A F-4 HI VIB**	**CCW TRAIN A SURGE TK LO-LO LVL**	**CCW TRAIN A LO HDR PRESS**
B	**NSCW TRAIN A LO HDR PRESS**	**NSCW TRAIN A TRANSF PMP LO DISCH PRESS**			**CCW TRAIN A SURGE TK HI/LO LVL**	**CCW TRAIN A LO FLOW**
C	**NSCW TRAIN A CLG TWR BASIN HI/LO LVL**		**NSCW TRAIN A DG CLR LO FLOW**	**NSCW TRAIN A RHR PMP & MTR CLR LO FLOW**	**CCW TRAIN A SURGE TK MAKE UP LVL**	**CCW TRAIN A RHR HX HI FLOW**
D		**NSCW TRAIN A CNMT CLR 1 & 2 LO FLOW**	**NSCW INTERTIE TRN A TO TRN B HI FLOW**			**CCW TRAIN A RHR HX LO FLOW**
E		**NSCW TRAIN A CNMT CLR 5 & 6 LO FLOW**	**NSCW TRAIN A NORM/BYP VLV MISPOSITIONED**	**RMWST VAC DEGASIFIER PNL ALARM**	**CCW TRAIN A RHR PMP SEAL LO FLOW**	**PRIMARY EQUIPMENT HI TEMP**
F		**NSCW TRN A RX CVTY CLG COIL LOW FLOW**	**RX MAKE UP STOR TK LO-LO LVL**	**RX MAKE UP STOR TK HI/LO LVL**		

ALB 36

	(01)	(02)	(03)	(04)
A	4160V SWGR 1AA02 TROUBLE	480V SWGR 1AB04 TROUBLE	480V SWGR 1AB05 TROUBLE	SEQ A TROUBLE
B	4160V SWGR 1AA02 NEG PH SEQ BUS PT	480V SWGR 1AB15 TROUBLE	SEQ A PNL DOORS OPEN	SEQ A SAFETY EQUIP FAILED TO START
C	480V MCC 1ABA TROUBLE	480V MCC 1ABB TROUBLE	480V MCC 1ABC TROUBLE	RAT FDR BRKR TRN A FAILED TO OPEN
D	480V MCC 1ABD TROUBLE	480V MCC 1ABE TROUBLE	480V MCC 1ABF TROUBLE	SEQ A IN MANUAL TEST
E	4160/480V SWGR TRN A TRANSFER SW ON LOCAL	BAT 1BD1B BRKR OPEN	BAT 1CD1B BRKR OPEN	BAT 1DD1B BRKR OPEN
F		BAT 1AD1B BRKR OPEN	ISO DEVICE PNL TRN A QIP1 TROUBLE	ISO DEVICE PNL TRN C QIP3 TROUBLE



NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DRAWINGS LISTED ON IX3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. CIRCUIT AFTER THE ISOLATION DEVICE IS CONSIDERED NON-CLASS 1E.
4. SEQUENCER START CONTACT CLOSES DURING SAFETY INJECTION (SI) OR LOSS OF OFFSITE POWER (LOP).
5. V=1NCQPCG.
6. J2=1NCQIP101. JUMPER WIRES FOR COMMON ANNUNCIATOR WINDOW ARE LOCATED IN 1NCQIP101. SEE E/D 1X3D-BD-L01E FOR PULLED CABLE TO ANNUNCIATOR.
7. HARD BATTERY POSITIVE SHALL NOT BE CONNECTED TO A CONTACT IN ONE DECK THAT IS ADJACENT TO A CONTACT ON THE SAME DECK TO WHICH HARD BATTERY NEGATIVE IS CONNECTED.

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	DESCRIPTION	REV
150/151M	W	CO-5	ST* 1456C05A04	1X4DB-136,137	P & I DIAGRAM	—
150/151GM	W	CO-9	ST* 1456C05A05	1X3D-AA-D02A	ONE LINE DIAGRAM	—
174	GE	HFA	12HFA51A42H	1X5DN091-1,2,92-1,125-2	LOGIC DIAGRAM	—
198,AX	GE	HMA	12HMA11B6	N/A	BECHTEL V/P	—
162-1	AGA	E7000	E7012PD			—
162-2	AGA	E7000	E7012PD			—
1852AX	GE	HFA	12HFA51A42H			
186M	ELECTRO-SWITCH	LOR	7802D			

NUCLEAR SAFETY RELATED A

COMPONEBT COOLING WATER PUMP MOTOR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
COMPONENT COOLING WATER SYSTEM
1-1203-P4-001-M01

NO.	VERSIONS	DATE	DR	CHK	APPV
12.0	REV PER ABN 1979006101E003,VER.1.0	3/15/05	JLO	TSL	DEW

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X3D-BD-L01A | VER. 12.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 009EK2.03 001/LOIT/RO/M/F 3.0/3.3/009EK2.03/LO-TA-37008///

Given the following:

- Unit 1 experienced a small break LOCA.
- 19012-C, "Post-LOCA Cooldown and Depressurization," is in progress.
- RCS pressure is 1315 psig and stable.
- Containment pressure is 3.5 psig.

Which one of the following completes the following statement?

19012-C requires a **minimum** SG NR level of __(1)__,

and

decay heat removal is accomplished by __(2)__.

	__(1)__	__(2)__
A.✓	10%	natural circulation and break flow
B.	10%	break flow ONLY
C.	32%	natural circulation and break flow
D.	32%	break flow ONLY

K/A

009 **Small Break LOCA**

EK2.03 **Knowledge of the interrelations between the small break LOCA and the following:**

S/Gs

K/A MATCH ANALYSIS

The question requires the candidate to recall the minimum required SG level during a small break LOCA and the reason this level is maintained. The cooling method is dependent on the SG level, which is interrelated with the specific LOCA conditions described in the stem.

EXPLANATION OF REQUIRED KNOWLEDGE

Per the Westinghouse background for E.1, during a small break LOCA RCS pressure will remain elevated even though enough subcooling to allow SI termination cannot be achieved. In this event, the RCS is cooled down and RCS pressure lowered until subcooling is achieved. In this configuration, break flow alone is not sufficient to provide RCS cooling. Therefore, SG level of at least 10% NR must be maintained to

provide a heat sink. This level ensures the SG tubes are covered on the secondary side. This level is increased to 32% NR in adverse containment conditions to compensate for instrument inaccuracies. The RCS is expected to be saturated or slightly subcooled. Therefore, SG tubes should remain covered.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per 19012-C step 9, with containment pressure below the adverse containment value, 10% NR level is required to ensure SG tubes are covered in order to achieve adequate cooling.

The second part is correct. With RCPs stopped and a small break LOCA in progress, per Westinghouse background, break flow alone is insufficient for core cooling. Single or two phase cooling flow from the SGs is required.

B. Incorrect. Plausible. The first part is correct. See the first part of Choice A above.

The second part is incorrect. As discussed above, break flow alone is insufficient for core cooling during a small break LOCA. However, break flow is a heat transfer method available for a larger LOCA where SG tubes are not filled. If a candidate is not able to properly assess the RCS conditions, this method could be assumed to be possible.

C. Incorrect. Plausible. The first part is incorrect. Containment pressure is <3.8 psig, which is below the adverse containment value. However, if containment pressure was above the adverse containment value, 32% NR narrow range SG level would be required.

The second part is correct. See the second part of Choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of Choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	009EK2.03
Importance Rating:	3.0 / 3.3
Technical Reference:	E-1 Background Rev 2, 4/30/2005 19012-C Rev 33.3
References provided:	None
Learning Objective:	LO-LP-37111-02 State the effect of various size breaks on the primary system with respect to temperatures and pressures. LO-LP-34700-26 Describe the process of reflux boiling. LO-LP-37112-01 Using EOP 19012 as a guide, briefly describe how each step is accomplished. LO-TA-37008 Perform Post-LOCA Cooldown and Depressurization of the RCS per 19012-C
Question origin:	BANK - Vogtle 2010 HL-15R NRC - Question # 009EK2.03
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.7 / 45.7
Comments:	

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. Check if RHR Pumps should be stopped:	
a. RHR Pumps - ANY RUNNING WITH SUCTION ALIGNED TO RWST.	a. Go to Step 9.
b. RCS pressure:	b. Go to Step 9.
1) Greater than 300 psig.	
2) Stable or rising.	
c. Stop RHR Pumps taking suction from the RWST.	
***8. IF RCS pressure lowers in an uncontrolled manner to less than 300 psig, THEN restart RHR pumps.**	
***9. Check intact SG levels:**	
a. NR level - AT LEAST ONE GREATER THAN 10% [32% ADVERSE].	a. IF all SGs NR levels less than 10% [32% ADVERSE], THEN maintain total feed flow greater than 570 gpm.
b. Maintain NR levels between 10% [32% ADVERSE] and 65%.	
c. NR level - ANY RISING IN AN UNCONTROLLED MANNER.	c. Go to Step 10.
d. Go to 19030-C, E-3 STEAM GENERATOR TUBE RUPTURE.	

Breaks ≤ 3/8" equivalent diameter hole

Breaks in this range are considered to be leaks, rather than small LOCAs, since the normal charging system can maintain reactor coolant inventory so that RCS pressure and pressurizer level do not decrease. Very slight system depressurization may occur but no automatic trip or safety injection signal would be generated. The core will remain fully covered provided that the steam generators are available to remove energy, and makeup flow is continuously delivered to the RCS. If charging flow is not available, the RCS transient behavior would be similar to the response described for Category 2.

If the leak is within Technical Specification limits or it can be isolated, the plant could remain in power operation. If the leak is above Technical Specification limits and cannot be isolated, then the plant should go to a cold shutdown condition utilizing the normal shutdown procedures. During cooldown the charging system should maintain pressurizer level and the RCS depressurization should be controlled to conform to the normal cooldown limits.

Breaks 3/8" < diameter <~ 1", minimum safety injection, or Category 1 breaks above with no charging flow assumed

For these break sizes the normal makeup system cannot maintain level and pressure. The RCS will depressurize and an automatic reactor trip and safety injection signal will be generated. Provided that a secondary side heat sink exists, the RCS will reach an equilibrium pressure which corresponds to the pressure at which the liquid phase break flow equals the high pressure pumped safety injection flow. It has been verified that this equilibrium pressure condition will be established for plants with charging/SI pumps. This effect is described here by the presentation of a specific plant analysis for break sizes within this range. A general description of system behavior applicable to the sample transient is provided first, then specific comments concerning the sample analysis are provided.

Early in the transient a loss of subcooled liquid in the RCS occurs which results in a moderate depressurization to the pressure which corresponds to saturation pressure in the core and hot legs. At this point the upper head, upper plenum, hot legs, and core begin to experience some slight voiding, but more than enough liquid flow exists through the core to keep it covered and cooled. During this period of voiding, however, RCS depressurization occurs at a much slower rate than during the time when the entire system was subcooled. Eventually the RCS depressurizes to the point of the reactor trip signal. Immediately following reactor trip, the RCS rapidly depressurizes, since only a fraction of the heat previous to trip is now being transferred to the primary fluid. Due to this rapid depressurization following reactor trip, a safety injection signal is quickly generated. Within a few minutes of the reactor trip time, an equilibrium pressure is established which is above the steam generator pressure. The fluid conditions in the RCS at the time of equilibrium pressure establishment may be characterized by slight voiding in the core and upper plenum and hot legs, and saturated or slightly subcooled liquid in the cold legs. Core heat is removed through the steam generators by continuous single or two-phase natural circulation.

The primary mixture level in the steam generators does not drain for breaks of this size, and the core remains covered throughout the entire transient provided that SI is not interrupted. Once equilibrium pressure is established there is no further net loss of liquid volume in the RCS. The natural circulation heat removal mode continues until the time that the break can remove all the decay heat (1 day for a 1" break). Prior to this time, auxiliary feedwater is required to maintain the heat sink. Since the equilibrium pressure established is determined by means of a volume balance of SI flow and break flow, the ΔP and ΔT from primary to secondary side, together with the cold safety injection water, may provide a total heat sink greater than the decay heat generated and a cooling of the primary fluid can occur.

This effect is evidenced by the sample transient to be described below in which the fluid in the primary becomes subcooled after a long period of time at equilibrium pressure. The RCS response would be similar to that described here regardless of the break location since the RCS will not undergo a draining, and break flow is much less than the loop flow.

Abnormal indications should be present in the containment for this category of LOCA although the response will be slower and milder than for larger break size LOCAs. For example, containment pressure will probably not reach the containment High-1 pressure of approximately 10 percent of containment design pressure (4-5 psig).

Case A - Standard 4-loop type plant, one-inch equivalent diameter break in the cold leg. The shutoff head of the SI system is 2100 psia due to spilling line flow loss. Minimum safeguards safety injection is assumed, and loss of offsite power is assumed to occur at the reactor trip time. Therefore, the only means of venting steam on the secondary side is through the steam generator safety valves. Minimum auxiliary feedwater is assumed available one minute after the reactor trip time. The analytical model and all other analysis assumptions are in conformance with Appendix K criteria.

Case A presents a sample analysis for a break near the larger end of the break size spectrum of this behavior mode category. Figure 1 shows the RCS pressure transient for this case. The RCS pressure stabilizes slightly above the steam generator safety valve set pressure. Figures 2 and 3 show the safety injection and break flows which are both stabilized at a flowrate of approximately 45 1b/sec (~340 gpm). Figure 4 shows that the pressurizer empties at approximately 10 minutes and does not refill. The system remains in a stable condition with the core covered and decay heat being adequately removed. In this analysis, at approximately 14 hours into the transient, hot leg flow becomes subcooled, indicating a gradual cooling trend of the RCS primary fluid. It should be noted that the subcooling of the RCS for breaks in this category would be increased substantially by increased safety injection flow, e.g., if both trains

were operating.

1. 010A3.02 001/LOIT AND LOCT/RO/C/A 3.6/3.5/010A3.02/LO-TA-60050///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- PORV-455A starts leaking by.
- OATC observes pressurizer pressure indicating 2205 psig and slowly lowering.

Which one of the following completes the following statement?

With no operator action, the pressurizer backup heaters __(1)__ be energized,

and

if pressure lowers to 2185 psig __(2)__ receive a close signal.

	__(1)__	__(2)__
A.	will	PORVs ONLY
B.✓	will	both PORVs and block valves
C.	will NOT	PORVs ONLY
D.	will NOT	both PORVs and block valves

K/A

010 **Pressurizer Pressure Control**

A3.02 **Ability to monitor automatic operation of the PZR PCS, including:**

Pressurizer pressure

K/A MATCH ANALYSIS

The question tests the candidate's ability to monitor automatic operation of the pressurizer pressure control circuit, and determine if the heaters, PORVs, and block valves are operating properly for the given conditions.

EXPLANATION OF REQUIRED KNOWLEDGE

With PZR pressure at 2235 psig, PZR Master Controller demand is 25%. When 1PV-455B, Loop 4 Pressurizer Spray Valve, starts leaking by, PZR pressure will lower. As pressure lowers, the demand on the PZR Master Controller will decrease resulting in increased output from the proportional heaters (spray valves are already closed). PZR Master Controller demand will continue to decrease to 9.4%, when all PZR backup

heaters with handswitches in AUTO will energize. This occurs at a pressurizer pressure of 2210 psig since the system was at normal equalibrium before the tansient. Reference 18000-C Figure 1 for a pictoral representation of the control circuit. (Note: Figure 1 depicts a steady state circuit. As transients occur, the entire scale "slides". Depending on the transient, functions driven out of the master controller will occur at pressures different from those depicted on Figure 1. However, the conditions of this failure have been chosen to coincide with Figure 1 to ensure predictability by the candidate.)

ALB11-D02 PRZR CONTROL LO PRESS AND HEATERS ON is driven from of the PZR Master Controller output and corresponds to 9.4% demand (2210 psig). This alarm alerts the operator that PZR backup heaters should have energized. ALB12-D03 PRZR PRESS LO PORV BLOCK alerts the operator that PZR pressure is <2185 psig and therefore, PZR Master Controller demand should be at 0%.

Per ARP 17012-C and LOGIC drawings 1X6AA02-00230 & 00494, when 2/4 pressurizer pressure transmitter sense <2185 psig, both PORVs and both Block valves recieve a direct signal to close. Additionally, PORVs require a pilot pressure to open. With block valves closed, PORVs are incapable of opening even if a demand signal is recieved from another part of the control circuit. Block valves can be manually opened using their handswitch with pressurizer pressure <2185 psig, however the valves will close again as soon as the handswitch is released. COPS must be armed to keep the block valves open with pressurizer pressure <2185 psig.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. As described in the Explanation of Required Knowledge above, as pressurizer pressure lowers to less than 2210 psig, pressurizer backup heaters will energize.

The second part is incorrect. As described in the Explanation of Required Knowledge above, when pressurizer pressure lowers to <2185 psig on 2 of 4 transmitters, both PORVs and both block valves will receive a direct signal to close. However, a candidate with insufficient knowledge of the pressurizer control circuit would find it reasonable for ONLY the PORVs to close and block valves remain open to ensure PORV would remain available if needed for manual operation.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. As described in the Explanation of Required Knowledge above, when pressurizer pressure lowers to <2185 psig on 2 of 4 transmitters, both PORVs and both block valves will receive a direct signal to close.

C. Incorrect. Plausible. The first part is incorrect. As described in the Explanation of Required Knowledge above, as pressurizer pressure lowers to less than 2210 psig, pressurizer backup heaters will energize. However, a candidate with insufficient knowledge of the PZR pressure control circiut may not recognize the backup heater

setpoint and determine that only proportional heaters should be energized.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 010A3.02
Importance Rating: 3.6 / 3.5

Technical Reference: 18000-C Rev 5, page 5
17011-1 Rev 16.0, pages 31 & 32
17012-C Rev 21.0, pages 23 & 24
LOGIC 1X6AA02-00230, Rev 8.0
LOGIC 1X6AA02-00235, Rev 9.0
LOGIC 1X6AA02-00236, Rev 7.0
LOGIC 1X6AA02-00494, Rev 2.0

References provided: None

Learning Objective: LO-PP-16301-02 Describe the purpose of the following pressurizer components or auxiliaries:

a. Variable Heaters
b. Backup Heaters
c. Spray Valves
d. Bypass Spray Valves
e. PORVs
f. PORV Block Valves

LO-PP-16303-02 Describe how the response of pressurizer pressure control to the following failures:

e. stuck open spray valve

LO-TA-60050 Respond to a stuck Open PRZR Spray valve using AOP 18000-C

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.5

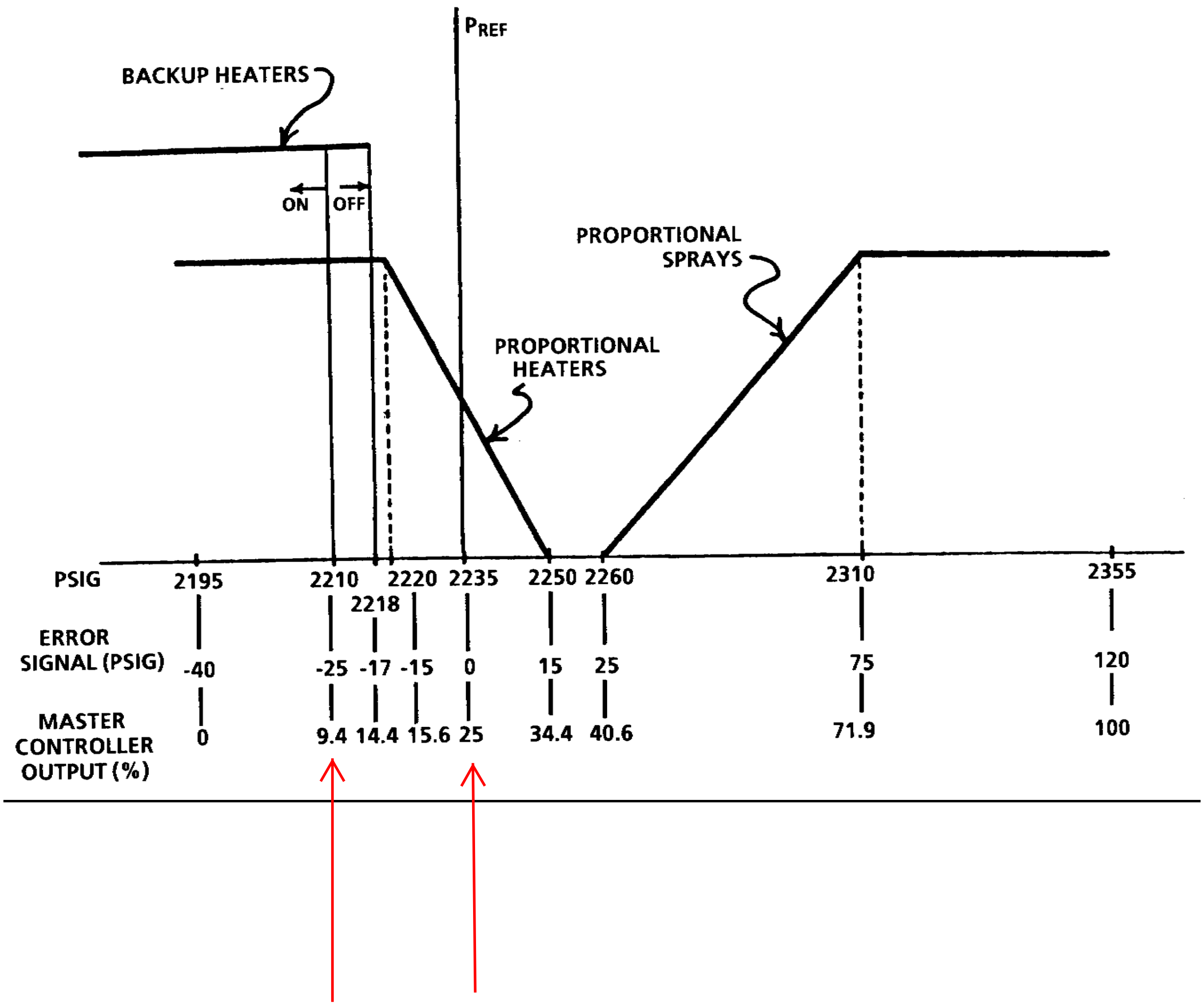
Comments:

You have completed the test!

FIGURE 1

Sheet 1 of 1

PRESSURIZER PRESSURE CONTROLLER BAND



WINDOW D02

ORIGIN	SETPOINT	PRZR CONTROL LO PRESS AND HEATERS ON
1-PIC-0445A output from selected channel: 1-PT-0455 OR 1-PT-0457	2210 psig	

1.0 PROBABLE CAUSE

1. Pressurizer Pressure Control System malfunction.

2. Pressurizer Spray or Relief Valve malfunction.

2.0 AUTOMATIC ACTIONS

Pressurize Backup Heaters will energize.

3.0 INITIAL OPERATOR ACTIONS

Check pressurizer pressure indications:

- IF an instrument failure is indicated, **initiate** 18001-C, "Primary Systems Instrumentation Malfunction".

- IF a failed PRZR Spray Valve, Safety Valve or PORV is indicated, **initiate** 18000-C “Pressurizer Spray, Safety Or Relief Valve Malfunction”.

- AT 1965 psig and lowering **trip RX** and **go to** 19000-C, “E-0 REACTOR TRIP OR SAFETY INJECTION”.

WINDOW D02
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NOTE

A large load increase or transient may cause pressurizer pressure to decrease temporarily below the alarm setpoint.

1. IF no instrument failure is indicated and no other reason for pressure decrease is evident, **initiate** 18004-C, "Reactor Coolant System Leakage".

2. Refer to Technical Specification LCO 3.4.1.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, 1X6AX01-106, 1X6AU01-182, 168, PLS RER 92-190, 93-213

WINDOW D03

ORIGIN	SETPOINT	
Any 2 of the following: 1-PT-0455 1-PT-0456 1-PT-0457 1-PT-0458	2185 psig	PRZR PRESS LO PORV BLOCK

1.0 **PROBABLE CAUSE**

1. RCS pressure transient during plant startup or shutdown.
2. RCS Pressure Control Malfunction.

2.0 **AUTOMATIC ACTIONS**

PRZR PORV 455A and 456A BLOCK VALVES 1-HV-8000A and 1-HV-8000B are blocked from opening, and will close if open with their handswitches in the auto position.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Verify** Block Valves are closed.
2. **Verify** Pressurizer Pressure Control System is responding to restore pressure.
3. IF a failure has occurred in the Pressurizer Pressure Control System, **refer** to 18001-C, "Primary Systems Instrumentation Malfunction."
4. IF RCS pressure decrease is due to excessive steam demand, **check** Steam Dumps and Atmospheric Relief Valves closed.
5. IF equipment failure is indicated, **initiate** maintenance as required.

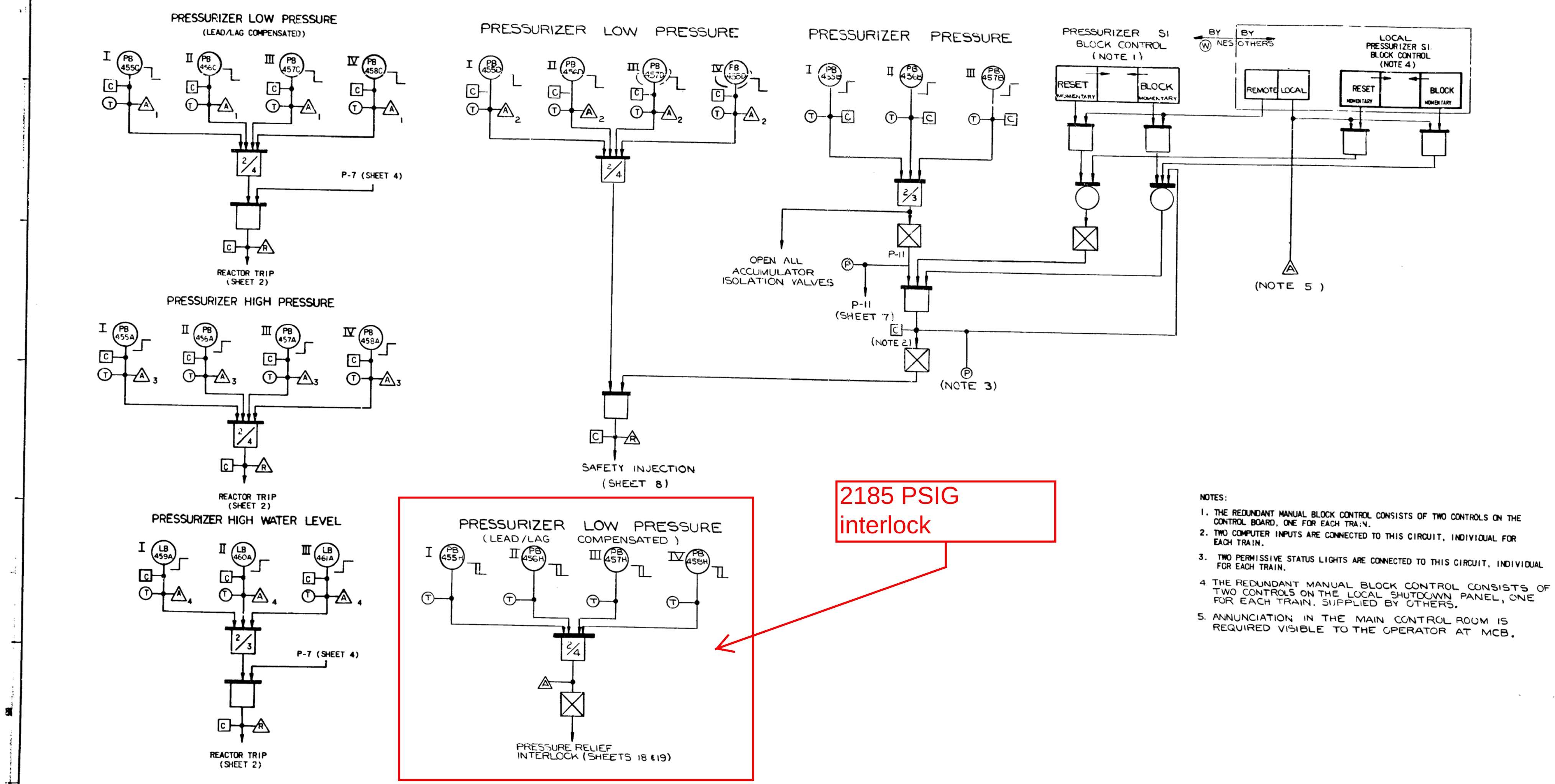
WINDOW D03
(Continued)

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, 1X6AA02-230, 1X6AU01-168; PLS



NOTES:

1. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
2. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT, INDIVIDUAL FOR EACH TRAIN.
3. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCUIT, INDIVIDUAL FOR EACH TRAIN.
4. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTROLS ON THE LOCAL SHUTDOWN PANEL, ONE FOR EACH TRAIN. SUPPLIED BY OTHERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED VISIBLE TO THE OPERATOR AT MCB.

GEORGIA POWER COMPANY
PLANT: ALVIN W. VOGTLE NUCLEAR PLANT
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-9146
AUTHORITY: J. L. VOTA
ENGR. LTR. NO. EP/SA-47348

Westinghouse Electric Corporation
TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRESSURIZER TRIP SIGNALS
7243D07
SHEET - 6

VOGTLE ELECTRIC GENERATING PLANT JOB NO. 9510
EQUIPMENT TAG NO. various
STARTUP DESIGNATION NO. various
ACTIVITY NO. various
SYSTEM NO. various
CATEGORY NO. NA
RETROFITTING REQUIRED ☐ YES ☒ NO
DISTRIBUTION TO: FOR: REVIEW INFO.
IDENTIFYING TITLE OF THIS DOCUMENT: Functional diag.
Bechtel Log No. X6AA02-230-8

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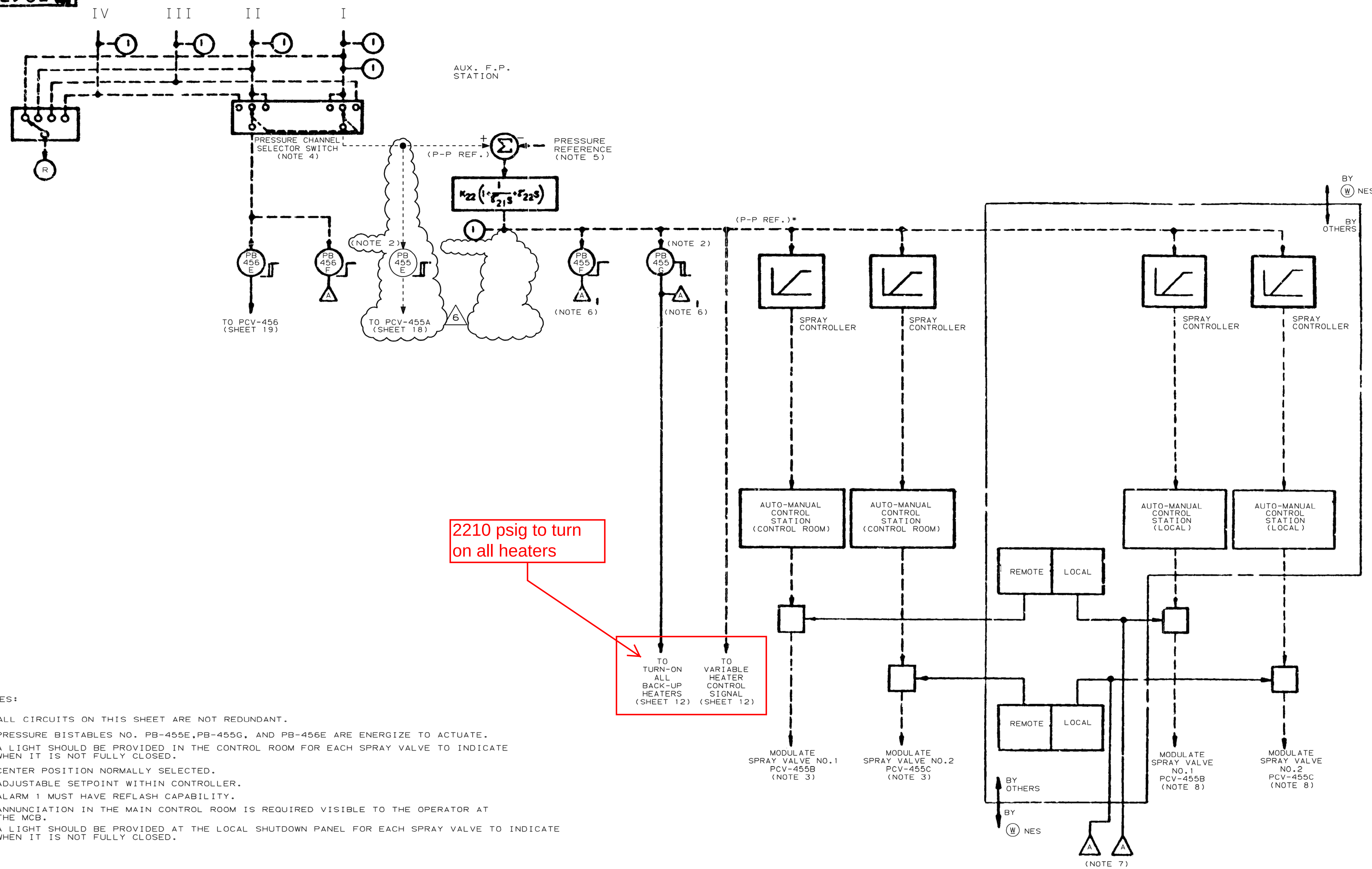
DATE RECEIVED 3-14-85
DATE 5/2/85

I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL

7243D07 SHEET-11

PRESSURIZER PRESSURE CHANNELS



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. PRESSURE BISTABLES NO. PB-455E,PB-455G, AND PB-456E ARE ENERGIZE TO ACTUATE.
3. A LIGHT SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN IT IS NOT FULLY CLOSED.
4. CENTER POSITION NORMALLY SELECTED.
5. ADJUSTABLE SETPOINT WITHIN CONTROLLER.
6. ALARM 1 MUST HAVE REFLASH CAPABILITY.
7. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED VISIBLE TO THE OPERATOR AT THE MCB.
8. A LIGHT SHOULD BE PROVIDED AT THE LOCAL SHUTDOWN PANEL FOR EACH SPRAY VALVE TO INDICATE WHEN IT IS NOT FULLY CLOSED.

NO.	REVISIONS	DATE	DR	CHK	APPV
6	INC. PER DCP 98-V1N0061	11-06-00	CD	EOG	MWD

SCS REVISIONS

DOCUMENT STATUS CODE 1

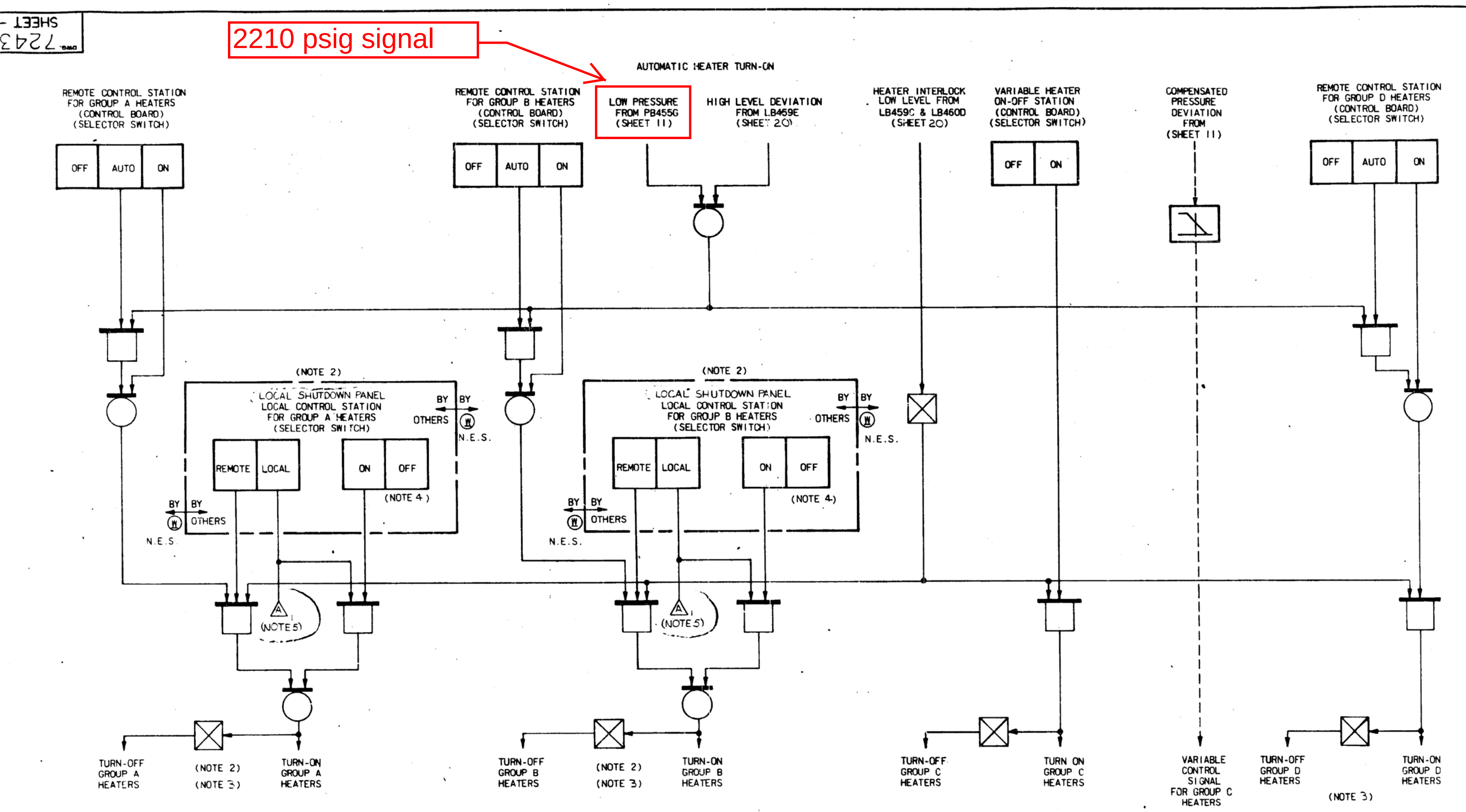
1X6AA02-00235-9

Westinghouse Electric Corporation

GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRESSURIZER PRESSURE CONTROL

7243D07
SHEET-11
SUB 6

SHEET - 12
7243D07



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. GROUP A AND GROUP B HEATERS MUST BE ON SEPARATE VITAL POWER SUPPLIES WITH THE LOCAL CONTROL SEPARATED SO THAT ANY SINGLE FAILURE DOES NOT DEFEAT BOTH.
3. BACK UP HEATER STATUS INDICATION IN CONTROL ROOM.
4. PRECAUTIONS SHOULD BE TAKEN TO AVOID MANUAL HEATER OPERATION, WHICH WOULD CAUSE HEATER DAMAGE, IF THE WATER LEVEL UNCOVERS THE HEATERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED TO BE VISIBLE TO THE OPERATOR AT THE MAIN CONTROL BOARD.

GEORGIA POWER COMPANY
PLANT: ALVIN W. VOGTLE NUCLEAR PLANT
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-5709
AUTHORITY: J. L. VOTA
ENGR. LTR. NO. EP/SA-36110

MICROFILM QUALITY ACCEPTABLE

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

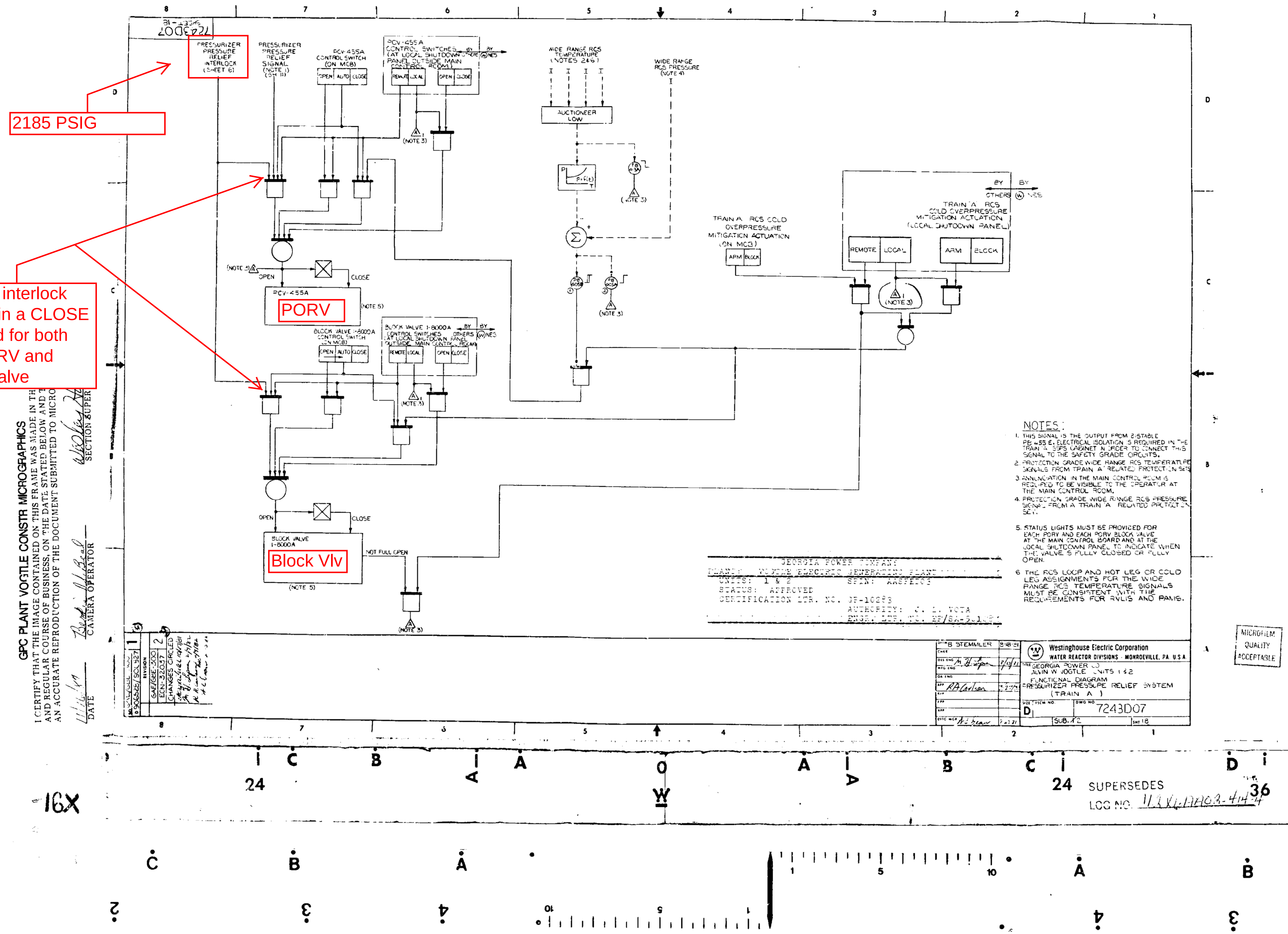
DFTM. P HEARN
DES. ENG.
APP.

TITLE: GEORGIA POWER CO.
ALVIN W VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAM
PRESSURIZER HEATER CONTROL

DIMENSIONS IN INCHES
DO NOT SCALE

7243D07
SHEET - 12
SUB 2345

Resub for micro film quality



VOGTLE ELECTRIC GENERATING PLANT — JOB NO. 9510

EQUIPMENT TAG NO. var

STARTUP DESIGNATION NO. var
ACTIVITY NO. var
SYSTEM NO. var
CATEGORY NO. NA

RETROFITTING REQUIRED ☐ YES ☒ NO

DISTRIBUTION TO: FOR: REVIEW	INFO.
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NSSS	
BOP	
CONTROL SYSTEMS	✓
ELECTRICAL	✓
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HAZARDS	
CIVIL/STRUCTURAL	
NUCLEAR	
STRESS/PLANT DESIGN	
CODES AND STANDARDS	
ARCHITECTURAL	
STARTUP	
CONSTRUCTION	
NOT REQ'D BY ENGRG	
CLIENT	
EQUIP. QUALIFICATION	
M & QS	
WESTINGHOUSE	

IDENTIFYING TITLE OF THIS DOCUMENT: functional diag.

Bechtel Log No 1 X6AA02-494-42 10-22-87

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DATE RECEIVED 1-9-86 SIGNED

DOCUMENT STATUS
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3☐ REVISE AND RESUBMIT WORK MAY NOT PROCEED
4☐ INFORMATION ONLY DISTRIBUTION REQ'D? ☐ YES
M☐ RESUBMIT – NOT ACCEPTABLE FOR MICROFILM WORK MAY PROCEED

DATE 10/27/87

PF-5987 (9510) 9-84

1. 010K6.04 001/LOIT/RO/C/A 2.9/3.2/010K6.04/LO-PP-16301-06//HL-15 NRC/

Given the following Unit 1 conditions:

- 1PV-455A, Pressurizer PORV, is stuck slightly open.
- 1HV-8000A, PORV Block Valve, will not close.
- 1PV-455A tail pipe temperature is reading ~280^oF.
- RCS pressure is stable at 1920 psig.
- Pressurizer vapor space temperature is 630^oF.
- PRT pressure is 35 psig and slowly rising.
- Containment pressure is 0 psig.

Which one of the following completes the following statement?

The 1PV-455A tail pipe temperature indication __(1)__ reading correctly,

and

with 1PV-455A partially open, its tail pipe temperature will rise to approximately __(2)__.

REFERENCE PROVIDED

	__(1)__	__(2)__
A.	is	630^oF and then stabilize
B.✓	is	338^oF and then lower to ~212^oF
C.	is NOT	630^oF and then stabilize
D.	is NOT	338^oF and then lower to ~212^oF

K/A

010 **Pressurizer Pressure Control**

K6.04 **Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS:**

- PRT

K/A MATCH ANALYSIS

This question requires the candidate to evaluate the tailpipe temperature of a stuck open PORV. The candidate is then required to determine the effect the rupture of the PRT rupture disk would have on the PORV tailpipe temperature.

EXPLANATION OF REQUIRED KNOWLEDGE

The candidate is required to convert PSIG to PSIA for use on a REFERENCE PROVIDED steam table/Mollier Diagram and compare the temperature of the current pressurizer and PRT conditions against the indicated tailpipe temperatures. Then, the candidate is required to recognize that as the event progresses, PRT pressure will slowly rise until the PRT rupture disk fails at 100 psig. At this time, PRT tailpipe discharge pressure will quickly lower to Containment Pressure.

Per the steam stables, 1935 psia and 630F correspond to a saturated system with an enthalpy of approximately 1140. With a PRT pressure of 50 psia, enthalpy of 1140 is saturated with a temperature of 281F. Therefore the indication is correct. As PRT pressure rises to approximately 115 psia, the system remains saturated and temperature will rise to 338F. As soon as the disk ruptures, pressure will quickly drop to 15 psia and remains saturated with a temperature of 212F.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per the Explanation of Required Knwoledge above, the given plant conditions will result in a tailpipe temperature of 281F.

The second part is incorrect. As discussed in the Explanation of Required Knowledge, the tailpipe temperature will rise to 338F and then quickly lower to 212F when the PRT ruptures. If the candidate does not understand the characteristics of the throttling process, it is reasonable to assume tailpipe temperature would eventually rise to equal the RCS temperature and remain there.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per the Explanation of Required Knwoledge above, the given plant conditions will result in PRT temperature rising to 338F and then lowering to 212F after the rupture disk blows.

C. Incorrect. Plausible. The first part is incorrect. As discussed in the Explanation of Required Knowledge above, the tailpipe temperature is expected to be 281F for the conditions listed. However, a candidate that does not understand throttling processes would find it reasonable for tailpipe temperature to be much higher than 281F.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	010K6.04
Importance Rating:	2.9 /3.2

Technical Reference: 2000 ASME Steam Tables

References provided: 2000 ASME Steam Tables

Learning Objective: LO-LP-34300-22 Explain the reduction of process pressure from throttling using an enthalpy-entropy (h-s) diagram or temperature-entropy (T-s) diagram.
LO-PP-16301-06 Determine the expected tail pipe temperature for a leaking or open PORV.

Question origin: MODIFIED - HL15 NRC Question # 007K1.01

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.7

Comments:

You have completed the test!

1. 011EK2.02 001/LOIT/RO/C/A 2.6/2.7/011EK2.02/LO-TA-37009///

Initial condition:

- Unit 1 experienced a reactor trip and SI.

Current conditions:

- The OATC aligns ECCS for Cold Leg Recirculation.
- RCS pressure is 1450 psig.
- 19010-C, "Loss of Reactor or Secondary Coolant," is in progress.

Which one of the following completes the following statements?

Prior to the LOCA, each RHR pump is aligned for injection into __(1)__ Cold Legs.

Following alignment to Cold Leg Recirculation, ECCS injection **flow** to the core is being provided by __(2)__.

	__(1)__	__(2)__
A.	2	CCPs ONLY
B.	2	CCPs and SIPs
C.	4	CCPs ONLY
D.✓	4	CCPs and SIPs

K/A

011 Large Break LOCA

EK2.02 Knowledge of the interrelations between the Pumps and the Large Break LOCA.

K/A MATCH ANALYSIS

The question requires the candidate to recall how many Cold Legs each RHR pump is capable of injecting into in standby alignment. Additionally, the candidate is required to determine which ECCS pumps will be injecting water into the core during the LOCA based on plant conditions.

EXPLANATION OF REQUIRED KNOWLEDGE

In standby alignment, each RHR pump is capable of injecting into 4 Cold Legs. The individual RHR pumps inject into 2 cold legs. However, the cross-connect valves HV8716A&B are open which allows each pump to inject into the other train cold legs also. During cold leg recirculation, HV-8716A&B are closed to prevent pump runout. Reference 1X4DB121 and 1X4DB122.

The approximate shutoff head of the ECCS pumps are as follows:

RHR - 300 psig
SI - 1625 psig
CCPs - 2700 psig

With RCS pressure at 1450 psig, RHR will be operating on mini-flow and the SI pumps and the CCPs will be injecting into the core.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Each RHR pump is capable of injecting into all 4 cold legs. However, each RHR feeds only two cold legs and the other train cold legs are fed via the cross-connect valves. A candidate with insufficient knowledge of RHR standby alignment who assumes HV-8716A&B are CLOSED will believe this answer correct.

The second part is incorrect. With RCS pressure at 1450 psig, the SIPs are operating below shutoff head and are injecting into the core along with the CCPs. A candidate unfamiliar with shutoff head pressure for an SIP may assume that the SIP is not injecting at this pressure.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. With RCS pressure at 1450 psig, the SIPs are operating below shutoff head and are injecting into the core along with the CCPs.

C. Incorrect. Plausible. The first part is correct. Each RHR pump is capable of injecting into all 4 cold legs - 2 cold legs from it's own piping and 2 cold legs via the cross-connect valves.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	011EK2.02
Importance Rating:	2.6 / 2.7

Technical Reference: EOP 19013-C Rev 29.2, page 5
19010-C Rev 34.3
1X4DB121, Rev 42.0
1X4DB121, Rev 51.0

References provided: None

Learning Objective:

LO-PP-13101-06 State the following parameters for the ECCS pumps:
- a. shutoff head (pressure)
- b. rated flow (gpm)
- c. design features to prevent runout and provide pump miniflow

LO-PP-13101-08 Explain the operation of ECCS in each of the three modes of operation.

LO-PP-13101-09 Describe the normal standby alignment of the ECCS.

LO-TA-13009 Manually align ECCS for Cold Leg Recirculation Phase using EOP 19013-C.

LO-TA-37009 Respond to a Large Break Loss of Primary Coolant per 19010-C.

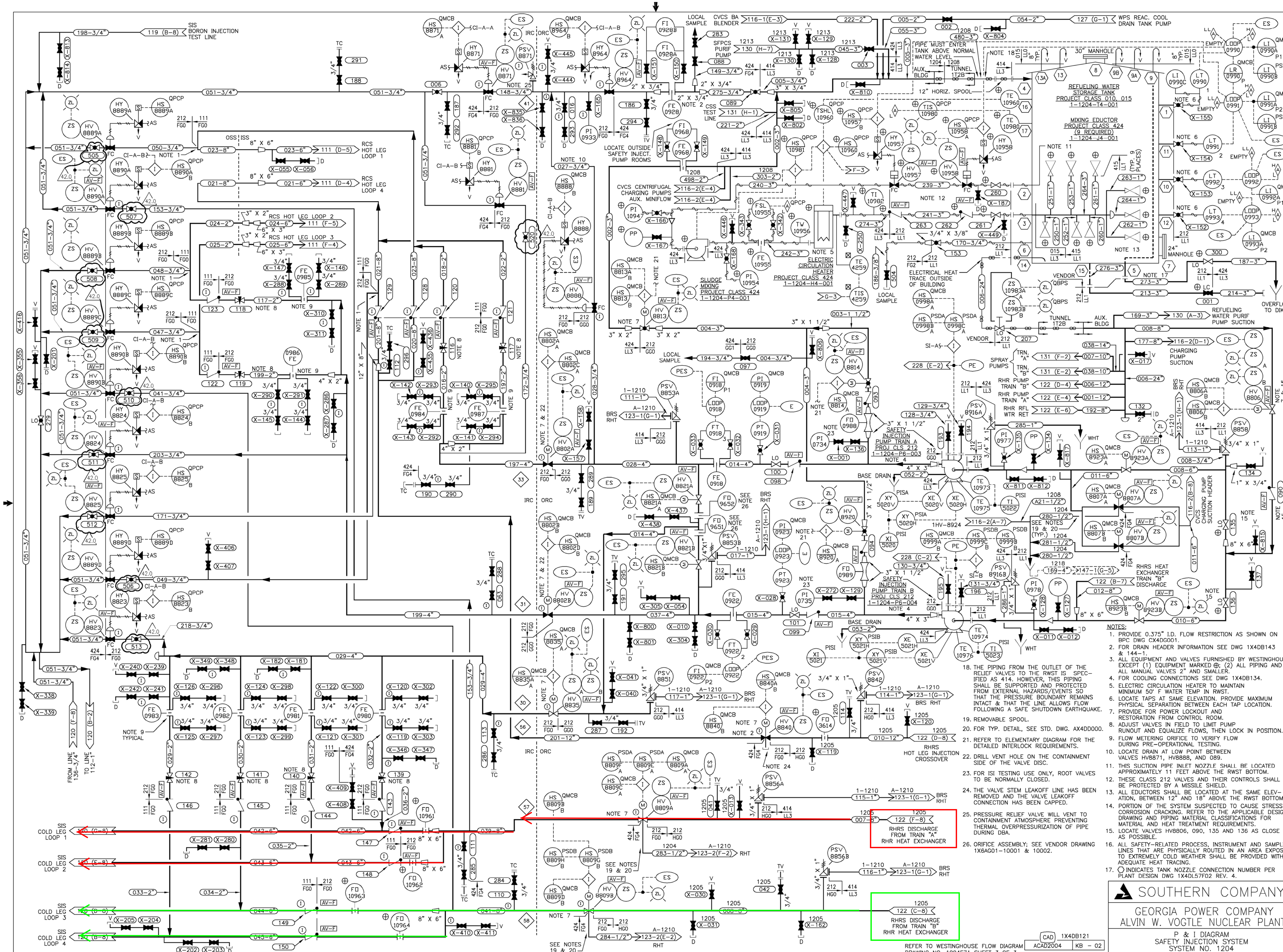
Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.7
Comments:

You have completed the test!

	ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
3.	Initiate ATTACHMENT A to align ECCS Pumps to the Cold Leg Recirculation flowpath and continue with Step 4.		
4.	Notify Health Physics that radiation levels in the Auxiliary Building will change when Cold Leg Recirculation is established.		
5.	Make a page announcement to clear personnel from the Auxiliary Building prior to initiating Cold Leg Recirculation.		
6.	Initiate Continuous Actions Page.		
*7.	**Check RWST level – GREATER THAN 8%.**	*7.	Stop any ECCS Pumps taking suction from the RWST.
*8.	**Check if SI pumps should be stopped.**		
	a. RCS pressure - GREATER THAN 1625 PSIG.		a. IF RCS pressure rises to greater than 1625 psig, THEN stop SI Pumps. Go To Step 9.
	b. Stop SI Pumps.		
9.	Check ATTACHMENT A - COMPLETE.	9.	Do NOT continue with this procedure until ATTACHMENT A has been COMPLETED.



NOTES:

1. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
2. FOR DRAIN HEADER INFORMATION SEE DWG 1X4DB143 & 144-1.
3. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT (1) EQUIPMENT MARKED ⊕; (2) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
4. FOR COOLING CONNECTIONS SEE DWG 1X4DB134.
5. ELECTRIC CIRCULATION HEATER TO MAINTAIN MINIMUM 50° F WATER TEMP IN RWST.
6. LOCATE TAPS AT SAME ELEVATION. PROVIDE MAXIMUM PHYSICAL SEPARATION BETWEEN EACH TAP LOCATION.
7. PROVIDE FOR POWER LOCKOUT AND RESTORATION FROM CONTROL ROOM.
8. ADJUST VALVES IN FIELD TO LIMIT PUMP RUNOUT AND EQUALIZE FLOWS, THEN LOCK IN POSITION.
9. FLOW METERING ORIFICE TO VERIFY FLOW DURING PRE-OPERATIONAL TESTING.
10. LOCATE DRAIN AT LOW POINT BETWEEN VALVES HV8871, HV8888, AND 089.
11. THIS SUCTION PIPE INLET NOZZLE SHALL BE LOCATED APPROXIMATELY 11 FEET ABOVE THE RWST BOTTOM.
12. THESE CLASS 212 VALVES AND THEIR CONTROLS SHALL BE PROTECTED BY A MISSILE SHIELD.
13. ALL EDUCTORS SHALL BE LOCATED AT THE SAME ELEVATION, BETWEEN 12" AND 18" ABOVE THE RWST BOTTOM.
14. PORTION OF THE SYSTEM SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
15. LOCATE VALVES HV8806, 090, 135 AND 136 AS CLOSE AS POSSIBLE.
16. ALL SAFETY-RELATED PROCESS, INSTRUMENT AND SAMPLING LINES THAT ARE PHYSICALLY ROUTED IN AN AREA EXPOSED TO EXTREMELY COLD WEATHER SHALL BE PROVIDED WITH ADEQUATE HEAT TRACING.
17. ⬡ INDICATES TANK NOZZLE CONNECTION NUMBER PER PLANT DESIGN DWG 1X4DL57F02 REV. 4.
18. THE PIPING FROM THE OUTLET OF THE RELIEF VALVES TO THE RWST IS SPECIFIED AS 414. HOWEVER, THIS PIPING SHALL BE SUPPORTED AND PROTECTED FROM EXTERNAL HAZARDS/EVENTS SO THAT THE PRESSURE BOUNDARY REMAINS INTACT & THAT THE LINE ALLOWS FLOW FOLLOWING A SAFE SHUTDOWN EARTHQUAKE.
19. REMOVABLE SPOOL.
20. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
21. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK REQUIREMENTS.
22. DRILL VENT HOLE ON THE CONTAINMENT SIDE OF THE VALVE DISC.
23. FOR ISI TESTING USE ONLY, ROOT VALVES TO BE NORMALLY CLOSED.
24. THE VALVE STEM LEAKOFF LINE HAS BEEN REMOVED AND THE VALVE LEAKOFF CONNECTION HAS BEEN CAPPED.
25. PRESSURE RELIEF VALVE WILL VENT TO CONTAINMENT ATMOSPHERE PREVENTING THERMAL OVERPRESSURIZATION OF PIPE DURING DBA.
26. ORIFICE ASSEMBLY; SEE VENDOR DRAWING 1X6AG01-10001 & 10002.

Q

REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E71 SHEET 3 OF 3.

CAD 1X4DB121
ACAD2004 KB - 02

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

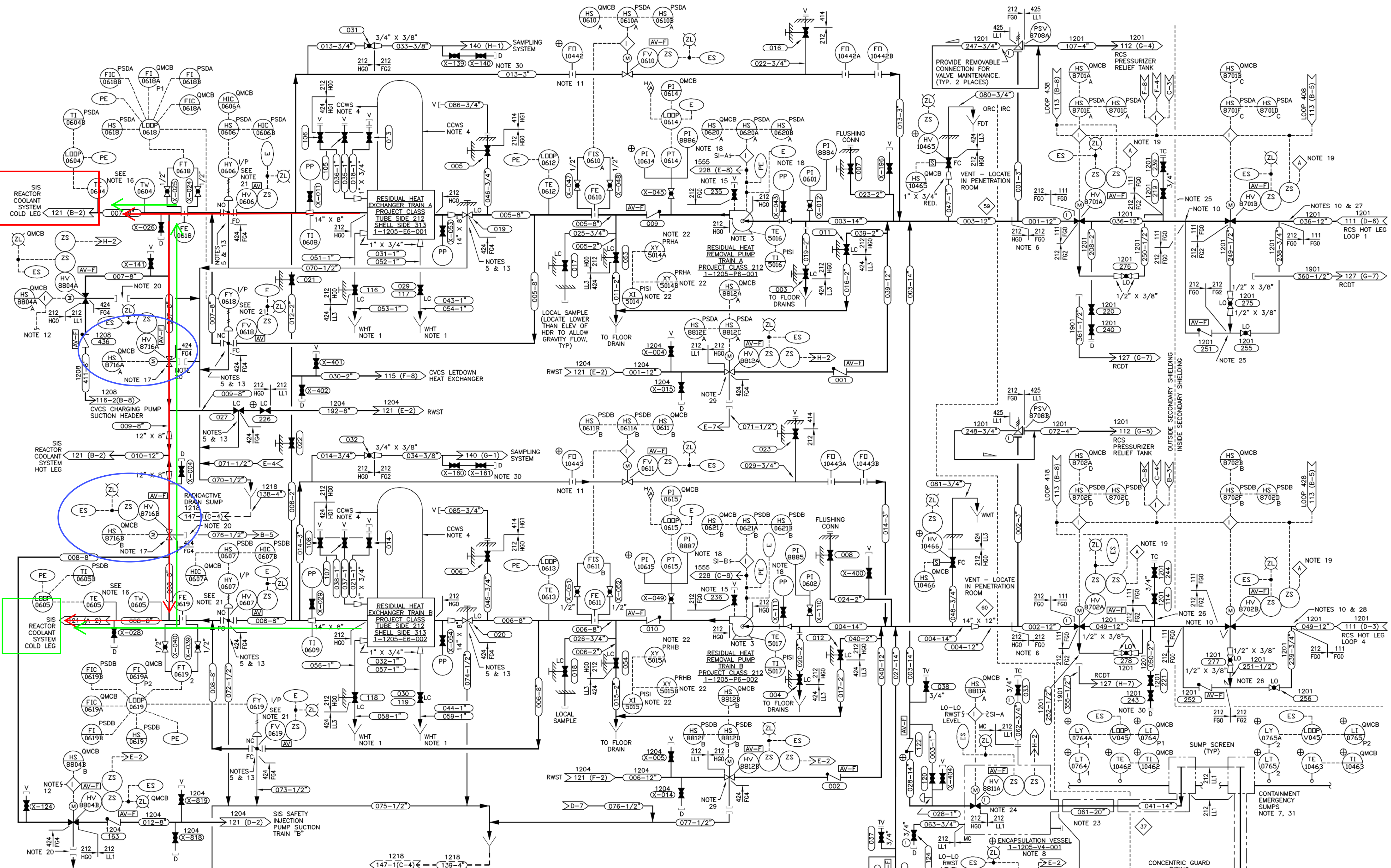
P & I DIAGRAM
SAFETY INJECTION SYSTEM
SYSTEM NO. 1204

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
42.0	REVISED PER ABN-1071073501M004, VER. 1.0	10/20/09	KB	CFC	JMR	X

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB121	42.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. FOR DRAIN HEADER INFORMATION SEE DWGS 1X4DB143, 144-1, 1X4DB145-5 AND 1X4DB183.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT: (1) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER, AND (2) EQUIPMENT MARKED ⊕.
3. FOR PUMP AND MOTOR COOLING WATER CONNECTIONS, SEE DWGS 1X4DB134 & 1X4DB137.
4. FOR HEAT EXCHANGER SHELL SIDE CONNECTIONS SEE DWG 1X4DB137.
5. PROVIDE 12" REMOVABLE SPOOL.
6. MAKE PIPING MATERIAL CLASS CHANGE AT FIRST WELD OF PENETRATION SLEEVE INSIDE THE CONTAINMENT.
7. INSTALL 150 #R.F. FLANGE ON INLET OF EACH 14"ø PIPE FOR PERIODIC TESTING. PROVIDE ONE MATING BLIND FLANGE TO BE DRILLED AND TAPPED FOR TEST CONNECTION. MATING FLANGE MUST BE REMOVED FOR NORMAL OPERATION.
8. VESSEL IS CONSTRUCTED TO SEISMIC CATEGORY 1 AND ASME III CLASS MC.
9. PORTIONS OF THIS SYSTEM ARE SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
10. PROVIDE 0.375" ID FLOW RESTRICTION AS SHOWN ON DWG CX4DG001. 51.0
11. THE ORIFICE SHOULD BE LOCATED ON A HORIZONTAL STRETCH OF PIPE.
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. FOR TYPICAL DETAIL SEE STD. DWG AX4DD000.
14. DELETED.
15. HIGH POINT VENT VALVE (BECHTEL-FURNISHED) FOR SEAL PIPING - SEE 1X6AF02-25.
16. TE IS A SURFACE TYPE CLAMP ON RTD, TW IS INSTALLED BUT NOT USED.
17. VENT HOLE IS DRILLED ON RESPECTIVE RHR PUMP SIDE OF DISC.
18. FOR ISI TESTING USE ONLY. INSTRUMENT ISOLATION VALVE TO BE NORMALLY CLOSED.
19. ALARM ON OPEN HOT LEG VALVE AND HIGH RCS PRESSURE.
20. DELETED.
21. AIR ISOLATED DURING POWER OPERATION TO ENSURE VALVE FAILS TO ITS SAFETY POSITION (FULL OPEN OR FULL CLOSED).
22. ABANDON IN-PLACE INSTRUMENTS XI5014, XY5014A, XY5014B, XI5015, XY5015A, AND XY5015B.
23. 4" VIEW PORT TO BE USED TO VERIFY MOTOR OPERATED VALVE STROKE TEST WITHOUT REMOVING THE VESSEL HEAD.
24. VENT HOLE IS DRILLED ON CONTAINMENT SIDE OF DISC.
25. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-249-01.
26. FLOW RESTRICTOR PROVIDED AS SHOWN ON 1J4-1201-251-01.
27. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-036-01.
28. 0.25 INCH FLOW RESTRICTOR AS SHOWN ON 1K4-1201-049-02.
29. A SECTION OF THE LEAKOFF LINE HAS BEEN REMOVED AND BOTH ENDS CAPPED.
30. THESE ARE KEROTEST VALVES AND MAY NOT ALLOW REVERSE FLOW.
31. SUMP SCREEN TAG NO'S. 1-1205-G2-001 (TRAIN "A") AND 1-1205-G2-002 (TRAIN "B")

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. NO. 1094E72

CAD 1X4DB122 ACAD2007 LPM - 07

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
51.0	REVISED PER ABN -V02177, VERSION 1.0	05/19/10	LPM	LCF	PML	X

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
RESIDUAL HEAT REMOVAL
SYSTEM NO. 1205

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB122	51.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 011K6.03 001/LOIT AND LOCT/RO/C/A 2.9/3.3/011K6.03/LO-TA-60030///

Initial conditions:

- Unit 1 is at 50% reactor power.
- RCS pressure is 2235 psig.
- Pressurizer level control is selected to CH 459 / 461.
- All pressurizer heaters are energized.

Current condition:

- Pressurizer level transmitter, 1LT-461, fails LOW.

- NO operator action has been taken.

Which one of the following completes the following statement?

1LV-460, Letdown Isolation Valve, __(1)__ close,

and

the pressurizer backup heaters are __(2)__.

	__(1)__	__(2)__
A.	will	energized
B.✓	will	de-energized
C.	will NOT	energized
D.	will NOT	de-energized

K/A

011 Pressurizer Level Control

K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS:

- Relationship between PZR level and PZR heater control circuit.

K/A MATCH ANALYSIS

The question tests the candidates knowledge of the impact on the pressurizer level control system from a control circuit failure - PZR level. From this the candidate must predict PZR heater system and spray valve response.

EXPLANATION OF REQUIRED KNOWLEDGE

Pressurizer level control uses primary and secondary control circuits. The level transmitters that feed the control circuits are selectable - as the question is written, LT-459 is primary and LT-461 is secondary. The primary channel uses the level input to control pressurizer level. There are interlocks with the pressurizer pressure control circuit to energize heaters on a 5% level deviation and to de-energize heaters and isolate letdown if level is <17%. The secondary channel is a backup to the primary and will also de-energize heaters and isolate letdown if level is <17%. The primary channel closes LV-459 and letdown orifice valves. The secondary channel closes LV-460 and letdown orifice valves.

If LT-461 (secondary) fails low, heaters are de-energized and letdown isolates. As pressurizer level increases due to loss of letdown, the primary channel will sense a level rise on LT-459 and lower charging flow. With charging in automatic, FIC-121 will close until charging flow is 42 gpm. The control circuit is limited to ensure minimum seal injection flow can be maintained with some flow remaining through the charging nozzle to prevent thermal transients on the nozzle welds (42 gpm total charging flow with 32 gpm seal injection flow and 10 gpm flow through the charging nozzle). Pressurizer level control also utilizes auctioneered high Tavg to provide a program level. Pressurizer level program is varied from 25% to 60% from 557F to 588.4F.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. With 459/461 selected for control, LT-461 is the input for the secondary channel. When the secondary channel indicates <17%, letdown will isolate. LV-460 will close and LV-459 will remain open. The letdown orifice isolation valves will close.

The second part is incorrect. With 459/461 selected for control, LT-461 is the input for the secondary channel. When the secondary channel indicates <17%, pressurizer heaters de-energize. However, pressurizer pressure control operates the pressuirzer heaters only from the primary channel. A candidate with insufficient knowledge of the PRZ level control circuit may assume only the primary channel controls the pressurizer heaters. As such, heaters would assume to be unaffected by LV-461. Therefore, this distractor is plausible.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. With 459/461 selected for control, LT-461 is the input for the secondary channel. When the secondary channel indicates <17%, pressurizer heaters de-energize.

C. Incorrect. Plausible. The first part is incorrect. With 459/461 selected for control, LT-461 is the input for the secondary channel. When the secondary channel indicates <17%, letdown will isolate. LV-460 will close and LV-459 will remain open. However, the primary channel is always an odd numbered instrument. Since LT-462 is calibrated for cold-cal use, LT-461 is used for both primary

and secondary control functions depending on selector switch position. As such, it is possible for a candidate to assume that LV-459 and LV-460 are controlled by LT-459 and LT-460 and not the selected primary and secondary circuits and conclude that LV-460 would remain open.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	011K6.03
Importance Rating:	2.9 / 3.3
Technical Reference:	ARP 17011-1 Rev 16, pages 14&15 LOGIC 1X6AA02-00236, Rev 7.0 LOGIC 1X6AA02-00496, Rev 1.0
References provided:	None
Learning Objective:	LO-PP-16302-02 Describe how the response of pressurizer level control to the following failures: a. controlling (primary & secondary) channel fails low b. controlling (primary & secondary) channel fails high c. controller high or low failure d. controlling channel fails as is LO-TA-60030 Respond to a Failure of Pressurizer Level Instrumentation per 18001-C
Question origin:	BANK
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7 /45.7
Comments:	

You have completed the test!

WINDOW B01

ORIGIN	SETPOINT
1-LT-0459 1-LT-0460 1-LT-0461	17%

PRZR LO LEVEL
HTR CNTL OFF
LTDN SECURED

1.0 **PROBABLE CAUSE**

1. Pressurizer level Control System Malfunction.
2. Charging - Letdown System Malfunction.
3. RCS cooldown.
4. Reactor Coolant System leak.

2.0 **AUTOMATIC ACTIONS**

1. All Pressurizer Heaters turn off.
2. Letdown isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** pressurizer level instrumentation.
2. IF instrument malfunction is indicated **Go To** 18001-C, "Primary Systems Instrumentation Malfunctions".
3. IF a Reactor Coolant System leak is indicated, **Go To** 18004-C, "Reactor Coolant System Leakage".

WINDOW B01
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. IF actual pressurizer level is low,

   a. **Verify** level is returning to normal, and

   b. IF necessary, **verify** Charging Line Flow Control Valve is open and **start** additional Charging Pumps as needed to increase level to normal.

2. After alarm clears, **restore** heaters and letdown in accordance with 13006-1, "CVCS Startup And Normal Operation".

5.0 **COMPENSATORY OPERATOR ACTIONS**

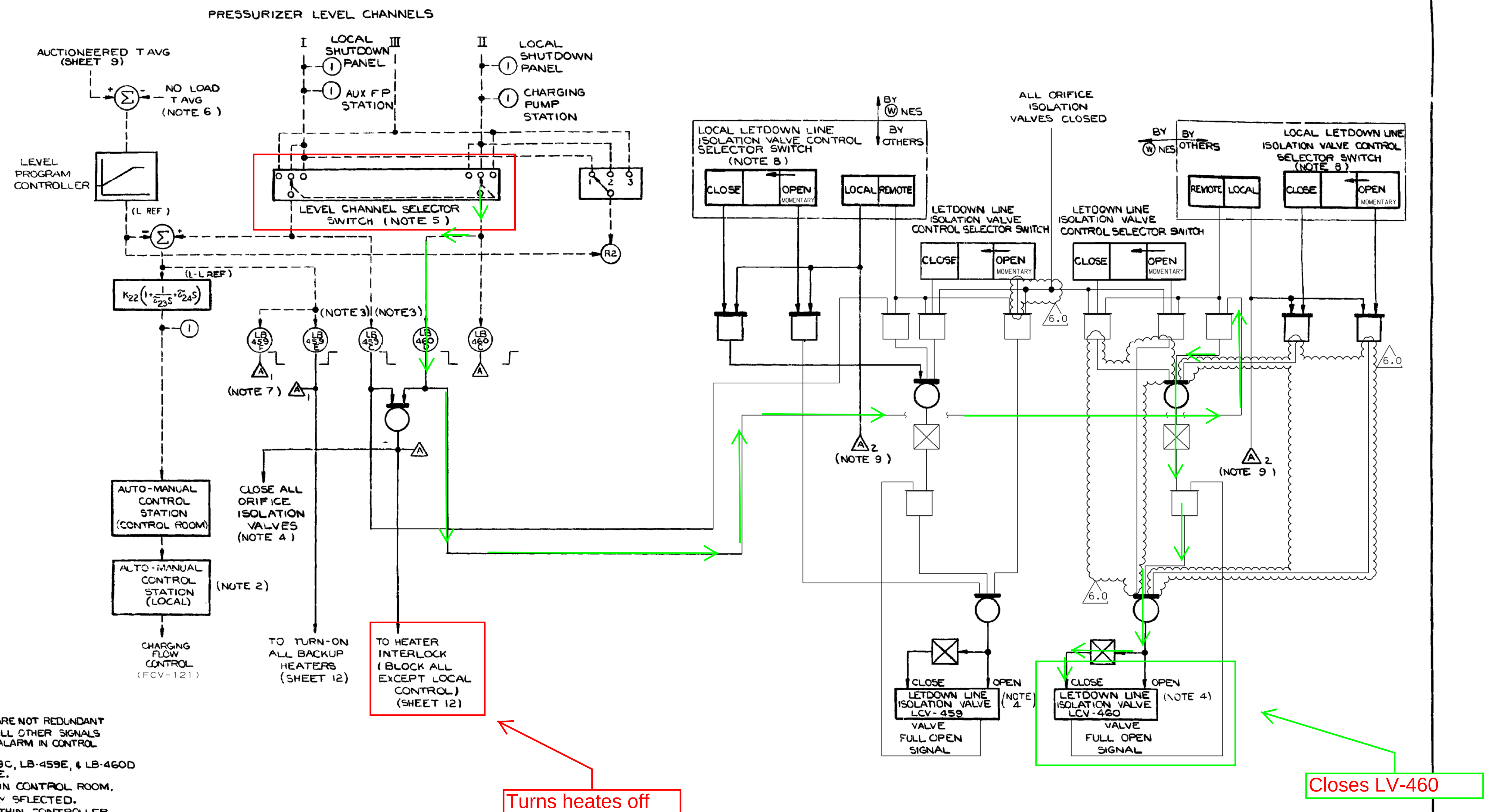
NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB112, 116-1, 1X6AU01-167, 183, PLS

7243D07 SHEET 20

PRESSURIZER LEVEL CHANNELS



NOTES

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM
3. LEVEL BISTABLES NO LB-459C, LB-459E, & LB-460D ARE ENERGIZE TO ACTUATE.
4. OPEN / SHUT INDICATION IN CONTROL ROOM.
5. CENTER POSITION NORMALLY SELECTED.
6. ADJUSTABLE SETPOINT WITHIN CONTROLLER.
7. ALARM I MUST HAVE RESET CAPABILITY.
8. MANUAL CONTROL OF THESE VALVES CONSISTS OF TWO SELECTOR SWITCHES ON THE LOCAL SHUTDOWN PANEL, ONE SWITCH PER VALVE. SUPPLIED BY OTHERS.
9. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED VISIBLE TO THE OPERATOR AT THE MCB

1	SO GAE/GBE 300 D 306-261/3CX8271		
2	GAE/GBE-300 ECN-32037 CHANGES CIRCLED	2/8/82	
LT	CHANGE		

DFTM B STEMMLER

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U S A

GEORGIA POWER CO
ALVIN W VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRESSURIZER LEVEL CONTROL

7243D07 SHEET 20

DO NOT SCALE

SIZE D 34x22

THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
TAB/SECT. N/A
PAGE N/A
FIGURE N/A
VERSION 6.0 DATE 12-21-05
REVISED BY SNC PER ABN-81187, VER.1.0
SEE MICROFILM FOR PREVIOUS REV. SIGNATURES.

BY	CHK'D	APPR.1	APPR.2
BgT	PWG	ASK	

CAD MICROSTATION BgT/01

1X6AA02-00496.DGN

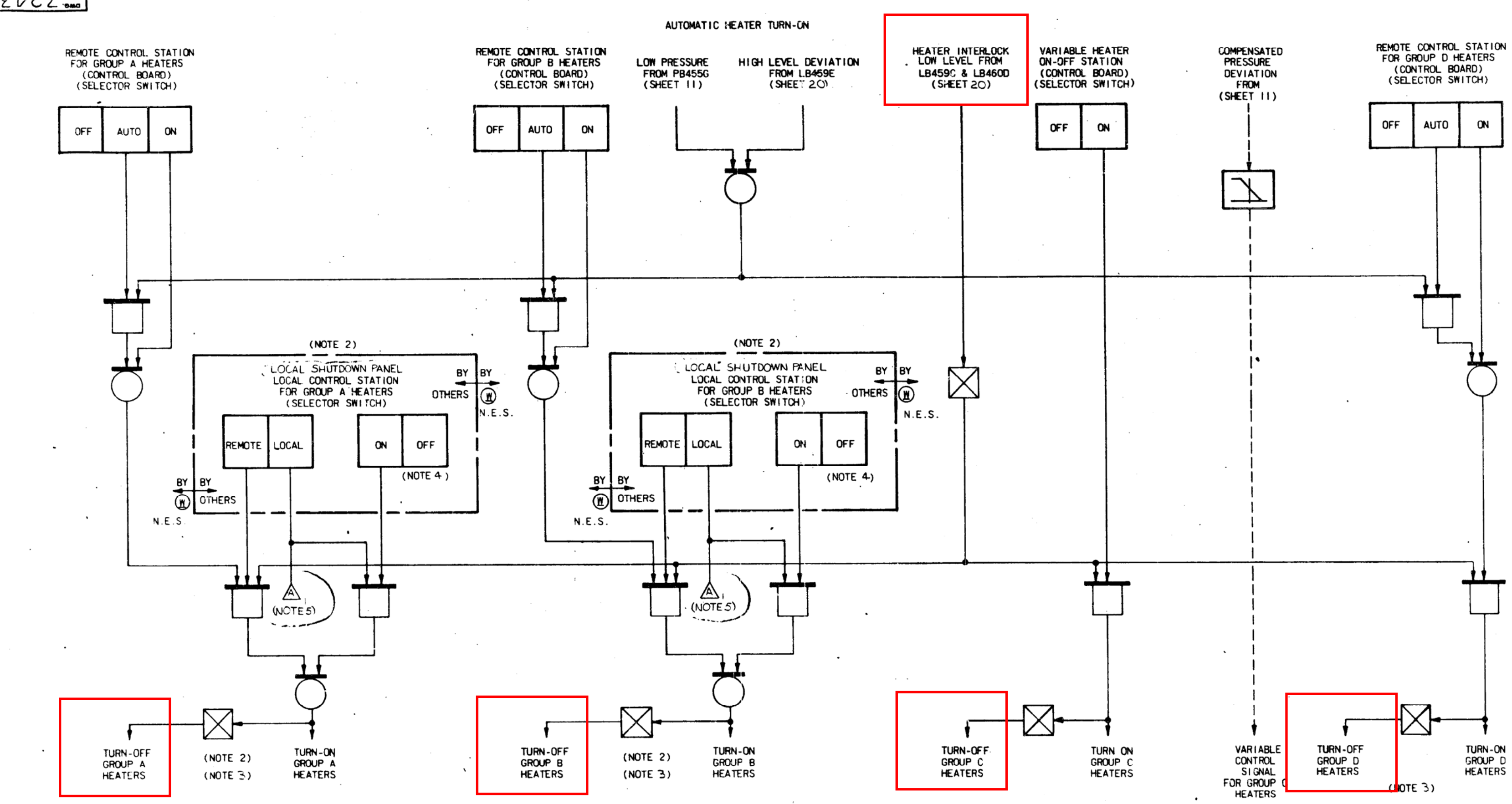
Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT UNIT NO. 1

TITLE: FUNCTIONAL DIAGRAM

VENDOR: WESTINGHOUSE P.O.#:

DRAWING NO. 1X6AA02-00496

DWG. 7243D07 SHEET - 12



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. GROUP A AND GROUP B HEATERS MUST BE ON SEPARATE VITAL POWER SUPPLIES WITH THE LOCAL CONTROL SEPARATED SO THAT ANY SINGLE FAILURE DOES NOT DEFEAT BOTH.
3. BACK UP HEATER STATUS INDICATION IN CONTROL ROOM.
4. PRECAUTIONS SHOULD BE TAKEN TO AVOID MANUAL HEATER OPERATION, WHICH WOULD CAUSE HEATER DAMAGE, IF THE WATER LEVEL UNCOVERS THE HEATERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED TO BE VISIBLE TO THE OPERATOR AT THE MAIN CONTROL BOARD.

GEORGIA POWER COMPANY
PLANT: ALVIN W. VOGTLE NUCLEAR PLANT
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-5709
AUTHORITY: J. L. VOTA
ENGR. LTR. NO. EP/SA-36110

MICROFILM QUALITY ACCEPTABLE

Resub for micro film quality

DFTM.	P HEARN
CHKR.	
DES. ENG.	7/12/73
MFG. ENG.	
MTLS. ENG.	
APP.	
APP.	
APP.	
DFTG. SUPV.	

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.
TITLE: GEORGIA POWER CO. ALVIN W VOGTLE UNITS 1 & 2 FUNCTIONAL DIAGRAM PRESSURIZER HEATER CONTROL
SCALE
DIMENSIONS IN INCHES
DO NOT SCALE
7243D07 SHEET - 12
SUB y 2345

I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL AND REGULAR COURSE OF BUSINESS ON THE DATE STATED BELOW AND THAT IT IS AN ACCURATE REPRODUCTION OF THE DOCUMENT SUBMITTED TO MICROGRAPHICS.
DATE: CAMERA OPERATOR SECTION SUPERVISOR

1. 012K5.01 001/LOIT/RO/M/F 3.3/3.8/012K5.01/LO-TA-60014///

Given the following:

- Unit 1 is at 100% reactor power.

Which one of the following completes the following statement?

The __(1)__ Reactor Trip protects against Departure from Nucleate Boiling (DNB),

and

the Trip Setpoint __(2)__ in response to **lowering** pressurizer pressure.

	__(1)__	__(2)__
A.	OT delta T	remains unchanged
B.✔	OT delta T	decreases
C.	OP delta T	remains unchanged
D.	OP delta T	decreases

K/A

001 Reactor Protection

K5.01 Knowledge of the operational implications of the following concepts as the apply to the RPS:

- DNB

K/A MATCH ANALYSIS

The question requires the candidate to understand the operational impact of lowering pressurizer pressure on Overtemperature Delta-T (OTdT) and the interrelationship between this reactor trip setpoint and DNB.

EXPLANATION OF REQUIRED KNOWLEDGE

Reactor trip for Overtemperature Delta-T (OTdT) utilizes narrow range RCS Tavg, pressurizer pressure, and delta-NI power to calculate a reactor trip setpoint, which is compared to the actual delta-T for each RCS loop. As RCS pressure lowers, subcooling lowers and the RCS approaches saturation and gets closer to DNB. Therefore, the OTdT setpoint lowers, getting closer to actual delta-T. If 2/4 loop OTdT setpoints are exceeded, an automatic reactor trip is initiated by SSPS. The calculation for the OTdT setpoint is listed in TS 3.3.1, Table 3.3.1-1 page 7 of 9. Candidates are required to understand the general effect of RCS parameter changes on the OTdT setpoint. Candidates are NOT required to calculate the setpoints. Per TS Bases for

3.3.1 Fu 6, the OTdT trip function is provided to ensure that the design limit DNBR is met.

Correspondingly, the Overpower Delta T (OPdT) reactor trip utilizes narrow range Tavg and the rate of change of Tavg to determine the setpoint (Delta-NI is nulled out). This trip protects against overpower in the fuel (Kw/ft). These two trips are routinely confused by candidates because the names are so similar. The trip bases and setpoint inputs are commonly jumbled and mismatched by candidates.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per TS Bases 3.3.1 Fu 6, the Overtemperature Delta T trip is designed to ensure DNBR is met.

The second part is incorrect. As RCS pressure lowers, subcooling lowers and the RCS approaches saturation and gets closer to DNB. Therefore, the OTdT setpoint lowers. However, candidates routinely confuse OPdT and OTdT. If the candidate did confuse OPdT and OTdT, the candidate would not expect the setpoint to change because pressure is not an input to OPdT.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. As RCS pressure lowers, subcooling lowers and the RCS approaches saturation and gets closer to DNB. Therefore, the OTdT setpoint lowers.

C. Incorrect. Plausible. The first part is incorrect. Per TS Bases 3.3.1 Fu 6, the Overtemperature Delta T trip is designed to ensure DNBR is met. However, candidates routinely confuse OTdT and OPdT (which protects from overpower in the fuel (Kw/ft)).

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	012K5.01
Importance Rating:	3.3 / 3.8

Technical Reference: TS 3.3.1, Ammendment No. 165, pages 3.3.1-20&21
TS Bases 3.3.1, Rev 1-8/02, pages B 3.3.1-16 thru 20

References provided: None

Learning Objective:

- LO-PP-16303-03 State the reactor trips and SI actuation signals, including set points and coincidences associated with pressurizer pressure protection channels.
- LO-PP-28103-03 List all reactor trip set points, coincidences, permissives, and blocks.
- LO-PP-28103-04 Discuss the bases for each reactor trip signal.
- LO-PP-56101-19 Describe what happens on a reactor trip and what is being protected by the trip.
- LO-TA-60014 Respond to Reactor Coolant System Leakage per 18004-C
- LO-TA-60029 Respond to a Failure of Pressurizer Pressure Instrumentation per 18001-C

Question origin: BANK

Cognitive Level: M/F

10 CFR Part 55 Content: 41.5 / 45.7

Comments:

You have completed the test!

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T

The Allowable Value of each input to the Overtemperature Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1) 0.5% ΔT span for the ΔT channel
(2) 0.5% ΔT span for the T_{avg} channel
(3) 0.5% ΔT span for the pressurizer pressure channel
(4) 0.5% ΔT span for the f_1(AFD) channel

$$\left[100\frac{\Delta T}{\Delta T_0}\frac{\{1+\tau_1 s\}}{\{1+\tau_2 s\}}\frac{1}{\{1+\tau_3 s\}}\right] \leq \left[K_1 - K_2\frac{\{1+\tau_4 s\}}{\{1+\tau_5 s\}}\left[T\frac{1}{\{1+\tau_6 s\}} - T'\right]^{(p)} - K_3\{P' - P\} - f_1(\mathrm{AFD})\right]$$

Where:	ΔT	measured loop specific RCS differential temperature, degrees F
	ΔT_0	indicated loop specific RCS differential at RTP, degrees F
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	lead-lag compensator on measured differential temperature
	τ_1, τ_2	time constants utilized in lead-lag compensator for differential temperature: τ_1 = 0 seconds, τ_2 = 0 seconds
	$\frac{1}{1+\tau_3 s}$	lag compensator on measured differential temperature
	τ_3	time constant utilized in lag compensator for differential temperature, ≤ 6 seconds
	K_1	fundamental setpoint, ≤ 114.9% RTP
	K_2	modifier for temperature, = 2.24% RTP per degree F
	$\frac{1+\tau_4 s}{1+\tau_5 s}$	lead-lag compensator on dynamic temperature compensation
	τ_4, τ_5	time constants utilized in lead-lag compensator for temperature compensation: τ_4 ≥ 28 seconds, τ_5 ≤ 4 seconds
	T	measured loop specific RCS average temperature, degrees F
	$\frac{1}{1+\tau_6 s}$	lag compensator on measured average temperature
	τ_6	time constant utilized in lag compensator for average temperature, ≤ 6 seconds
	T′	indicated loop specific RCS average temperature at RTP, ≤ 588.4 degrees F
	K_3	modifier for pressure, = 0.177% RTP per psig
	P	measured RCS pressurizer pressure, psig
	P′	reference pressure, ≥ 2235 psig
	s	Laplace transform variable, inverse seconds

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T (continued)

f_1(AFD) modifier for Axial Flux Difference (AFD):

1. for AFD between -23% and +10%, = 0% RTP
2. for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP
3. for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP

(p) The compensated temperature difference $\frac{\{1 + \tau_4 s\}}{\{1 + \tau_5 s\}}\left[T\frac{1}{\{1 + \tau_6 s\}} - T'\right]$ shall be no more negative than 3 degrees F.

Note 2: Overpower Delta-T

The Allowable Value of each input to the Overpower Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1) 0.5% ΔT span for the ΔT channel
(2) 0.5% ΔT span for the T_{avg} channel

$$\left[100\frac{\Delta T}{\Delta T_0}\frac{\{1 + \tau_1 s\}}{\{1 + \tau_2 s\}}\frac{1}{\{1+\tau_3 s\}}\right] \le \left[K_4 - \left[K_5\frac{\{\tau_7 s\}}{\{1 + \tau_7 s\}}\frac{1}{\{1+\tau_6 s\}}T\right] - K_6\left[T\frac{1}{\{1 + \tau_6\}} - T''\right] - f_2(\mathrm{AFD})\right]$$

Where:

ΔT	measured loop specific RCS differential temperature, degrees F
ΔT_0	indicated loop specific RCS differential at RTP, degrees F
$\frac{1+\tau_1 s}{1+\tau_2 s}$	lead-lag compensator on measured differential temperature
τ_1, τ_2	time constants utilized in lead-lag compensator for differential temperature: τ_1 = 0 seconds, τ_2 = 0 seconds
$\frac{1}{1+\tau_3 s}$	lag compensator on measured differential temperature
τ_3	time constant utilized in lag compensator for differential temperature, ≤ 6 seconds
K_4	fundamental setpoint, ≤ 110% RTP
K_5	modifier for temperature change: ≥ 2% RTP per degree F for increasing temperature, ≥ 0% RTP per degree F for decreasing temperature
$\frac{\tau_7 s}{1+\tau_7 s}$	rate-lag compensator on dynamic temperature compensation
τ_7	time constant utilized in rate-lag compensator for temperature compensation, ≥ 10 seconds
T	measured loop specific RCS average temperature, degrees F
$\frac{1}{1+\tau_6 s}$	lag compensator on measured average temperature

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A) ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux — High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature — including dynamic compensation for RTD response time delays.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. This results in a two-out-of-four trip logic. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

subcritical, boron dilution (see LCO 3.3.8) and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux — Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the Source Range Neutron Flux trip is blocked.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the Rod Control System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. Source range detectors also function to monitor for high flux at shutdown. This function is addressed in Specification 3.3.8. Requirements for the source range detectors in MODE 6 are addressed in LCO 3.9.3.

6. Overtemperature ΔT

The Overtemperature ΔT trip Function (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A) is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;

- pressurizer pressure — the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution — f(AFD)x, the f(AFD) Function is used in the calculation of the Overtemperature ΔT trip. It is a function of the indicated difference between the upper and lower NIS power range detectors. This Function measures the axial power distribution. The Overtemperature ΔT Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for RTD response time delays.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. A trip occurs if Overtemperature ΔT is indicated in two loops. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

This results in a two-out-of-four trip logic. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

Delta-T_0, as used in the overtemperature and overpower ΔT trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., differences in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Therefore, loop specific ΔT_0 values are measured as needed to ensure they represent actual core conditions.

The parameter K_1 is the principal setpoint gain, since it defines the function offset. The parameters K_2 and K_3 define the temperature gain and pressure gain, respectively. The values for T' and P' are key reference parameters corresponding directly to plant safety analyses initial conditions assumptions for the Overtemperature ΔT function. For the purposes of performing a CHANNEL CALIBRATION, the values for K_1, K_2, K_3, T', and P' are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T', the value for the hottest RCS loop will be set

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABLITY

6. Overtemperature ΔT (continued)

as close as possible to 588.4° F. The instrument uncertainty calculations and safety analyses, in combination, have accounted for loop variation in loop specific, full power, indicated ΔT and $T_{avg.}$ With respect to T_{avg}, a value for T' common to all four loops is permissible within the limits identified in the uncertainty calculations. Outside of those limits, the value of T' will be set appropriately to reflect indicated, loop specific, full power values. In the case of decreasing temperature, the compensated temperature difference shall be no more negative than 3 °F to limit the increase in the setpoint during cooldown transients. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overtemperature ΔT instrument channel is performing in a manner consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the magnitude of instrument drift from the as-left condition is within limits, and that the input parameters to the trip function are within the appropriate calibration tolerances for the defined calibration conditions (Ref. 7).

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

(continued)

1. 013K6.01 001/LOIT/RO/M/F 2.7/3.1/013K6.01/LO-TA-28013//HL18 NRC/016K1.09

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- The bistable(s) for Containment Pressure Channel I (1PT-937) are **de-energized**.
- No Tech Spec actions have been taken.

Which one of the following completes the following statement?

The MINIMUM number of ADDITIONAL channels required to initiate an actuation signal on High-1 is __(1)__,

and

the MINIMUM number of ADDITIONAL channels required to initiate an actuation signal on High-3 is __(2)__.

	__(1)__	__(2)__
A.	1	1
B.	1	2
C.	2	1
D.✓	2	2

K/A

013 **Engineered Safety Features Actuation System (ESFAS)**

K6.01 **Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS:**

- Sensors and detectors

K/A MATCH ANALYSIS:

The question addresses a de-energized Containment pressure channel. The candidate must determine how many of the remaining OPERABLE channels are required for High-1 and High-3 actuations to occur.

EXPLANATION OF REQUIRED KNOWLEDGE

ESFAS actuations associated with Containment Pressure are HI-1 for Safety Injection, HI-2 for Steam Line Isolation, and HI-3 for Containment Spray. HI-1 and HI-2 are

de-energize to actuation bistable. HI-3 is an energize to actuate bistable. HI-1 and HI-2 utilize a 2 of 3 logic utilizing Channel 2, 3, & 4 only. HI-3 is a 2 of 4 logic.

With Channel 1 PT-937 bistables de-energized, neither HI-1 nor HI-3 have any bistables tripped, HI-1 because Channel 1 is not used and HI-3 because they are energize to actuate bistables. Therefore, both HI-1 and HI-3 would require two additional bistables to trip before an actuation would occur.

The bullet stating "No Tech Spec actions have been taken" is required because Tech Spec actions would alter the state of the HI-3 Channel 1 bistable.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Channel 1 is not utilized for HI-1, and two channels are required for an actuation to occur. However, a candidate who does not have specific knowledge of which three containment pressure channels are utilized may find it reasonable for channels 1, 2, & 3 to be used. In that case, only one additional channel is required.

The second part is incorrect. HI-3 bistables are energize to actuate and two channels are required for an actuation to occur. However, a candidate without specific knowledge of the bistable behavior may think that the bistables are de-energize to actuate and determine that only one additional channel is required.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. HI-3 bistables are energize to actuate and two channels are required for an actuation to occur.

C. Incorrect. Plausible. The first part is correct. Channel 1 is not utilized for HI-1 and 2 channels are required for an actuation to occur.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The first part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	013K6.01
Importance Rating:	2.7 / 3.1

Technical Reference: TS 3.3.2, Amendment No. 165, pages 3.3.2-2 thru 3 & 9 thru 12
TS Bases 3.3.2, Rev 20, pages B 3.3.2-12, 13, 17, & 18
LOGIC 1X6AA02-00232, Rev 17.0
Picture of TSLB 4

References provided: None

Learning Objective:

LO-LP-39207-02 Given a set of Technical Specification and the Bases, determine for a specific set of plant conditions, equipment availability, and operational mode.

a. Whether any Tech Spec LCOs of section 3.3 are exceeded.
b. The required actions for any section 3.3 LCOs.

LO-LP-28103-05 List all ESFAS actuation signals with all applicable set points, coincidences, permissives, blocks, and discuss the system response to each ESF actuation signal.

LO-TA-28013 Trip Protection System Bistable

LO-TA-28014 Use the BTI panel to bypass a protection channel

Question origin: BANK - HL18 NRC Question # 013K6.01

Cognitive Level: M/F

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments:

You have completed the test!

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	--------------------NOTE------------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---	
	C.1 Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 Be in MODE 5.	60 hours
D. One channel inoperable.	--------------------NOTE------------------- A channel may be bypassed for up to 12 hours for surveillance testing. ---	
	D.1 Place channel in trip.	72 hours
	OR	
	D.2.1 Be in MODE 3.	78 hours
	AND	
	D.2.2 Be in MODE 4.	84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One Containment Pressure High-3 channel inoperable.	--------------------NOTE------------------ One additional channel may be bypassed for up to 12 hours for surveillance testing. --	
	E.1 Place channel in bypass.	72 hours
	OR	
	E.2.1 Be in MODE 3.	78 hours
	AND	
	E.2.2 Be in MODE 4.	84 hours
F. One channel inoperable.	F.1 Restore channel to OPERABLE status.	48 hours
	OR	
	F.2.1 Be in MODE 3.	54 hours
	AND	
	F.2.2 Be in MODE 4.	60 hours

(continued)

Table 3.3.2-1 (page 1 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4[(i)(j)] SR 3.3.2.7[(i)(j)] SR 3.3.2.8	≤ 4.4 psig	3.8 psig
d. Pressurizer Pressure - Low	1,2,3[(a)]	4	D	SR 3.3.2.1 SR 3.3.2.4[(i)(j)] SR 3.3.2.7[(i)(j)] SR 3.3.2.8	≥ 1856 psig	1870 psig
e. Steam Line Pressure - Low	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4[(i)(j)] SR 3.3.2.7[(i)(j)] SR 3.3.2.8	≥ 570[(b)] psig	585[(b)] psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.2-1 (page 2 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure						
High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≤ 22.4 psig	21.5 psig

(continued)

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

Table 3.3.2-1 (page 3 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Phase A Containment Isolation						
(a) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
(b) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
(c) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
4. Steam Line Isolation						
a. Manual Initiation	1,2(c),3(c)	2	F	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(c),3(c)	2	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

(c) Except when one main steam isolation valve and associated bypass isolation valve per steam line is closed.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure - High 2	1,2(c), 3(c)	3	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≤ 15.4 psig	14.5 psig
d. Steam Line Pressure						
(1) Low	1,2(c), 3(a)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≥ 570 (b) psig	585 (b) psig
(2) Negative Rate - High	3(d)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4(i)(j) SR 3.3.2.7(i)(j) SR 3.3.2.8	≤ 125 (e) psi/sec	100 (e) psi/sec

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

(c) Except when one main steam isolation valve and associated bypass isolation valve per steam line is closed.

(d) Below the P-11 (Pressurizer Pressure) interlock.

(e) Time constant utilized in the rate/lag controller is ≥ 50 seconds.

(i) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(j) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in NMP-ES-033-006, Vogtle Setpoint Uncertainty Methodology and Scaling Instructions.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Safety Injection - Automatic Actuation Logic and Actuation Relays (continued)

consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure — High 1

(PI-0934, PI-0935, PI-0936)

PI-0937 is not included.

NOTE: Containment pressure channels are also required OPERABLE by the Post Accident Monitoring Technical Specification.

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure — High 1 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

Thus, the high pressure Function will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties. Containment Pressure — High 1 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Safety Injection - Pressurizer Pressure — Low

This signal (PI-0455A, B, & C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458 & PI-0458A) provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the NTSP reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Containment Spray - Automatic Actuation Logic and Actuation Relays (continued)

this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation handswitches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray - Containment Pressure High — 3

(PI-0934, PI-0935, PI-0936, PI-0937)

NOTE: Containment Pressure Channels are also required OPERABLE by the Post Accident Monitoring Technical Specification.

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells and electronics) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Thus, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

This Function requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Containment Spray - Containment Pressure High — 3 (continued)

The Containment Pressure High-3 instrument Function consists of four channels in a two-out-of-four logic configuration. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure — High 3 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure — High 3 setpoints.

3. Phase A Containment Isolation

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic.

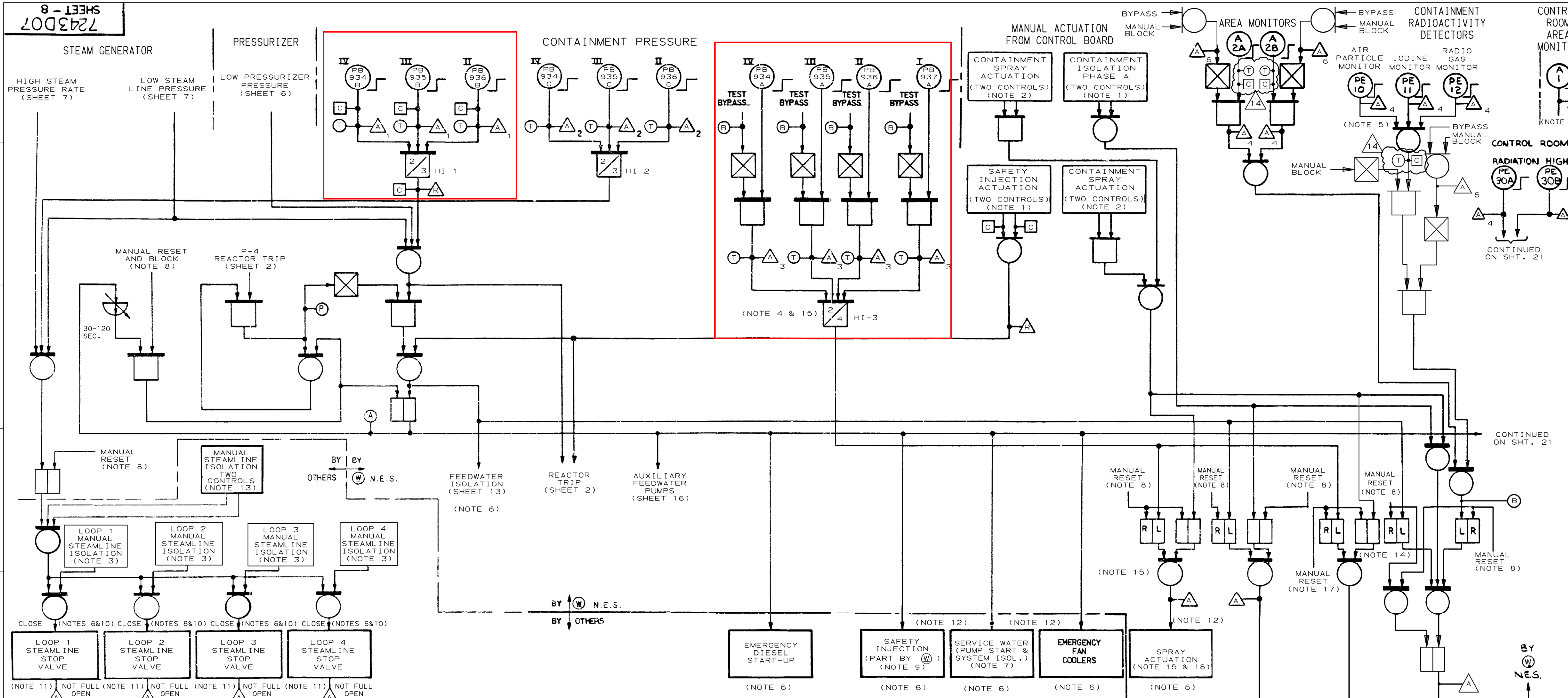
a. Phase A Isolation — Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

b. Phase A Isolation — Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Under specific conditions, a single inoperable actuation relay does not require that the affected automatic actuation logic function be

(continued)



NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, OPERATING EITHER CONTROL WILL ACTUATE.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS, ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE-TO-ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM, AND IS NOT SAFETY RELATED, HOWEVER, CIRCUITRY IS INPUT TO THE SAFEGUARDS SYSTEM THERFORE ISOLATION MUST BE PROVIDED.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
8. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
9. SAFETY INJECTION SEQUENCE REQUIREMENTS (IF SEQUENCING IS NECESSARY) ARE SPECIFIED BY W NUCLEAR ENERGY SYSTEMS.
10. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
11. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAM LINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF W NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE W SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE SEQUENCE TO LOAD THE CONTAINMENT SPRAY SYSTEM ON THE EMERGENCY DIESEL POWER SHOULD BE INDEPENDENT OF THE SEQUENCE TO LOAD THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY THE SEQUENCE INTERLOCK SHOULD BE SUCH THAT SPRAY WILL START WITHIN THE REQUIRED TIME INDEPENDENT OF THE SAFETY INJECTION SIGNAL STATUS.
17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.

NO.	REVISIONS	DATE	DR	CHK	APPV
14	INCORPORATED PER DCP 97-V1N0067	11-16-00	ELC	TSL	GLB

SCS REVISIONS

DOCUMENT STATUS CODE 1

1X6AA02-00232-17

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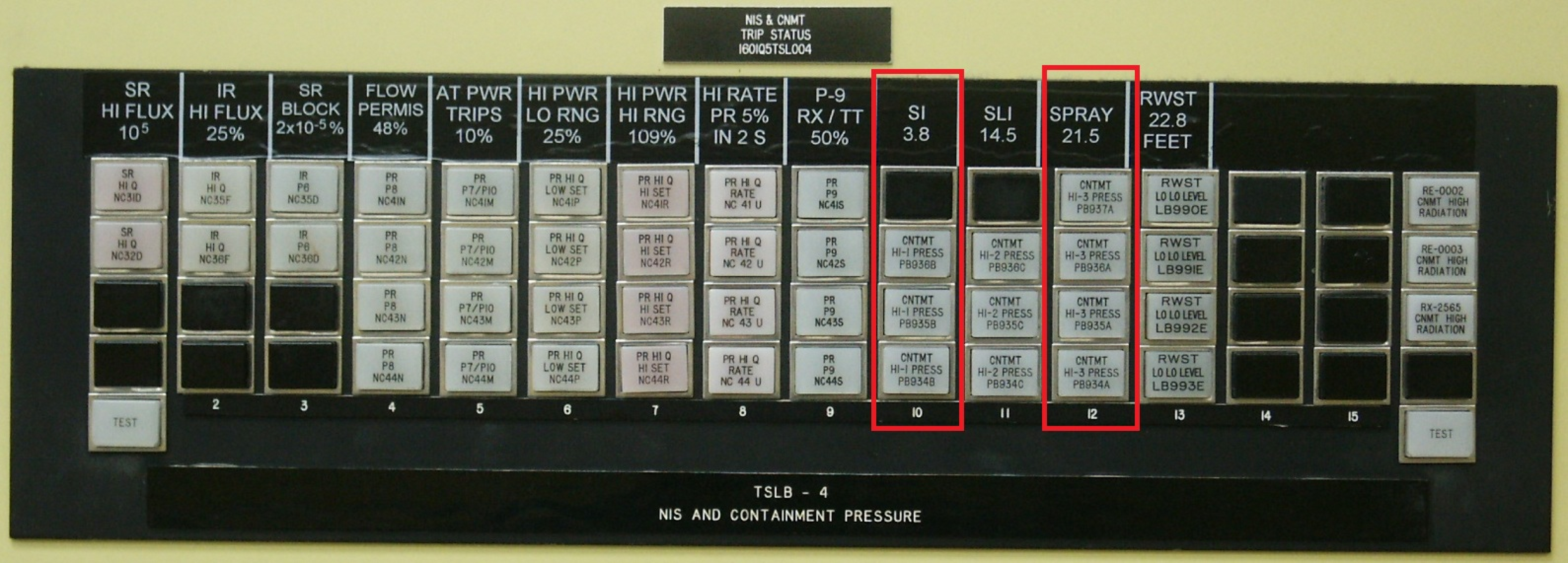
DFTM. P HEARN 5/10/73

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAM
SAFEGUARD ACTUATION SYSTEM

7243D07
SHEET - 8
SUB 13

DO NOT SCALE


NIS & CNMT
TRIP STATUS
I60IQ5TSL004
SR HI FLUX 10^5
IR HI FLUX 25%
SR BLOCK $2x10^{-5}$%
FLOW PERMIS 48%
AT PWR TRIPS 10%
HI PWR LO RNG 25%
HI PWR HI RNG 109%
HI RATE PR 5% IN 2 S
P-9 RX / TT 50%
SI 3.8
SLI 14.5
SPRAY 21.5
RWST 22.8 FEET
SR HI Q NC31D
SR HI Q NC32D
TEST
IR HI Q NC35F
IR HI Q NC36F
IR P6 NC35D
IR P6 NC36D
PR P8 NC41N
PR P8 NC42N
PR P8 NC43N
PR P8 NC44N
PR P7/P10 NC41M
PR P7/P10 NC42M
PR P7/P10 NC43M
PR P7/P10 NC44M
PR HI Q LOW SET NC41P
PR HI Q LOW SET NC42P
PR HI Q LOW SET NC43P
PR HI Q LOW SET NC44P
PR HI Q HI SET NC41R
PR HI Q HI SET NC42R
PR HI Q HI SET NC43R
PR HI Q HI SET NC44R
PR HI Q RATE NC 41 U
PR HI Q RATE NC 42 U
PR HI Q RATE NC 43 U
PR HI Q RATE NC 44 U
PR P9 NC41S
PR P9 NC42S
PR P9 NC43S
PR P9 NC44S
CNTMT HI-1 PRESS PB936B
CNTMT HI-1 PRESS PB935B
CNTMT HI-1 PRESS PB934B
CNTMT HI-2 PRESS PB936C
CNTMT HI-2 PRESS PB935C
CNTMT HI-2 PRESS PB934C
CNTMT HI-3 PRESS PB937A
CNTMT HI-3 PRESS PB936A
CNTMT HI-3 PRESS PB935A
CNTMT HI-3 PRESS PB934A
RWST LO LO LEVEL LB990E
RWST LO LO LEVEL LB991E
RWST LO LO LEVEL LB992E
RWST LO LO LEVEL LB993E
RE-0002 CNMT HIGH RADIATION
RE-0003 CNMT HIGH RADIATION
RX-2565 CNMT HIGH RADIATION
TEST
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11
12
13
14
15
TSLB - 4
NIS AND CONTAINMENT PRESSURE

1. 014A1.02 001/LOIT/RO/M/F 3.2/3.6/014A1.02/LO-PP-27201///

Initial condition:

- Unit 1 is at 100% reactor power and stable.

The following DRPI alarms and indications are observed:

- ALB10-D05 RPI URGENT ALARM is lit.
- ALB10-C05 RPI NON URGENT ALARM is lit.
- ALB10-E05 ROD AT BOTTOM is lit.
- Control rod H2 Rod Bottom LED is lit.
- Control rod H2 General Warning LED is flashing.
- Data A and Data B Failure LEDs are flashing.

Which one of the following completes the following statement?

Control rod H2 __(1)__,

and

automatic and manual rod motion __(2)__ inhibited.

	__(1)__	__(2)__
A.	dropped	is
B.	dropped	is NOT
C.	did NOT drop	is
D.✓	did NOT drop	is NOT

K/A

014 **Rod Position Indication System (RPIS)**

A1.02 **Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including:**

- Control rod position indication on control room panels.

K/A MATCH ANALYSIS

The question tests the candidate's ability to address multiple DRPI alarms and indications and determine if these conditions are indicative of an actual dropped rod or an instrumentation failure. In addition, the candidate must determine if there is any impact to the rod control system.

EXPLANATION OF REQUIRED KNOWLEDGE

All indications correspond to a simutaneous Data A and Data B failure for rod H2. The distinction between an actual dropped rod and a RPI failure are denoted by the presence of the General Warning LED and the Data A and Data B failure LEDs. Additionally, rod H2 is a low-worth control rod located on the outer edges of the core. If this rod dropped, little to no change in reactor power would be noticed. ALB10-E05 ROD AT BOTTOM is driven from the RPI position, and would alarm because of the indicated position of 0 steps on RPI.

The RPI system does not interface with the rod control system and therefore can not restrict rod motion. The RPI and Rod Control annunciators are commonly confused. The rod control urgent failure would send an inhibit signal to the PULSER/ OSCILLATOR and restrict rod motion in both auto and manual.

Reference ARP 17010-1 for specifics associated with expected plant conditions and responses.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The presence of the General Warning and Data A and Data B LEDs are symptoms of simulatenous RPI Data A and Data B failure and not a dropped rod. However, a candidate with insufficient knowledge of the RPI system would see the RPI rod at bottom LED and annunciator ALB10-E05 ROD AT BOTTOM and find it reasonable that the rod had actually dropped. These two conditions are entry symptoms for AOP 18003-C section A for a dropped rod in Mode 1. This is a common point of confusion among LOIT candidates.

The second part is incorrect. The RPI system does not interface with the rod control system and therefore can not restrict rod motion. However, the RPI and rod control alarms are commonly confused. If a candidate swaps the two systems and is thinking about a rod control urgent failure, then this answer would be correct.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. The RPI system does not interface with the rod control system and therefore can not restrict rod motion.

C. Incorrect. Plausible. The first part is correct. The presence of the General Warning and Data A and Data B LEDs are symptoms of simultaneous RPI Data A and Data B failure and not a dropped rod.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G2
K/A# 014A1.02
Importance Rating: 3.2 / 3.6

Technical Reference: ARP 17010-1, Rev 50.0, pages 3, 12-15, 21-24, 34-35, 44-46
AOP 18003-C, Rev 26.4, page 1

References provided: None

Learning Objective:

LO-PP-27101-20 Describe what occurs upon receipt of a Rod Control System urgent failure; include how rod motion is inhibited.

LO-PP-27201-02 Describe the principle of operation of the DRPI system.

LO-PP-27201-05 Describe how the DRPI system responds to a loss of Data A or B.

LO-PP-27201-06 State the conditons which will cause the following:

a. RPI Urgent Failure LEDs
b. RPI Urgent Failure annunciator
c. General Warning (GW) LED(s)
d. RPI Non-Urgent Failure annunciator
e. CCC Failure LEDs
f. Rod at bottom annunciator
g. Two or more rods at bottom annunciator
h. Rod Deviation annunicator
i. DRPI indication for a rod moves from 12 steps withdrawn to 18 steps withdrawn

Question origin: BANK

Cognitive Level: M/F

10 CFR Part 55 Content: 41.6 / 41.7 / 45.5

Comments:

You have completed the test!

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure provides instructions for malfunctions of the Rod Control System resulting in uncontrolled rod motion, dropped or misaligned rods.

SYMPTOMS
SECTION A, DROPPED RODS IN MODE 1

- ALB10-E5 ROD AT BOTTOM
- ALB10-F2 POWER RANGE HI NEUTRON FLX RATE ALERT
- ALB10-C2 POWER RANGE CHANNEL DEVIATION
- Rod bottom LED on digital rod position indication.
- Rod misaligned greater than 110 steps from demand position
- Tavg dropping.

SECTION B, UNCONTROLLED CONTINUOUS ROD MOTION IN ALL MODES

- Rod motion with invalid demand from the Automatic Rod Control System.
- Failure of rods to stop moving when the Rod Motion Switch is released.

SECTION C, MISALIGNED RODS IN MODE 1

- ALB10-C2 POWER RANGE CHANNEL DEVIATION
- ALB10-D2 POWER RANGE UP DET HI FLX DEV
- ALB10-E2 POWER RANGE LWR DET HI FLX DEV
- Failure of ALB10-C4 ROD BANK LO LIMIT or ALB10-D4 ROD BANK LO-LO LIMIT to reset during rod withdrawal.
- Rod misaligned greater than 12 steps and less than or equal to 110 steps from demand position.
- Quadrant power tilt ratio calculation exceeds 1.02.

ALB 10

	(1)	(2)	(3)	(4)	(5)	(6)
A	SR/IR SIG PROCESSOR TROUBLE	NIS SOURCE AND INTMD RANGE TRIP BYPASS	POWER RANGE HI NEUTRON FLX HI SETPOINT ALERT	REACTOR BYPASS BRKR BYA IN-OPERATE	REACTOR BYPASS BRKR BYA CLOSE	ROD CONTROL NON URGENT FAILURE
B	SOURCE RNG HI SHUTDOWN FLUX ALARM BLOCKED		POWER RANGE HI NEUTRON FLX LOW SETPOINT	REACTOR BYPASS BRKR BYB IN-OPERATE	REACTOR BYPASS BRKR BYB CLOSE	ROD CONTROL URGENT FAILURE
C	SOURCE RANGE HI FLUX LEVEL AT SHUTDOWN	POWER RANGE CHANNEL DEVIATION	OVERPOWER ΔT ROD BLOCK AND RUNBACK ALERT	ROD BANK LO LIMIT	RPI NON URGENT ALARM	NIS CHANNEL ON TEST
D	INTMD RANGE HI FLUX LEVEL ROD STOP	PWR RANGE UP DET HI FLX DEV	OVERPOWER ROD STOP	ROD BANK LO-LO LIMIT	RPI URGENT ALARM	ROD DEV
E	SR/IR REMOTE SIG PROCESSOR DPU-B TROUBLE	PWR RANGE LWR DET HI FLX DEV	OVERTEMP ΔT ROD BLOCK AND RUNBACK ALERT		ROD AT BOTTOM	RADIAL TILT
F	SR/IR AMPLIFIER TROUBLE	POWER RANGE HI NEUTRON FLX RATE ALERT		ROD DRIVE M-G SET TROUBLE	TWO OR MORE RODS AT BOTTOM	DELTA FLUX DEVIATION

WINDOW A06

ORIGIN	SETPOINT	
Power Cabinet Logic Cabinet	Not Applicable	ROD CONTROL NON URGENT FAILURE

1.0 **PROBABLE CAUSE**

1. Power Cabinet Non Urgent Failure:

   Loss of +28V DC power supply PS1 or PS2, or -24V DC power supply PS3 or PS4 due to low line voltage, blown fuse or failure of module's AC supply train.

2. Logic Cabinet Non Urgent Failure:

   a. Loss of 16.5V DC power supplies PS1, PS2, PS4, or PS5,

   b. Loss of 100V DC power supplies PS3 or PS6.

2.0 **AUTOMATIC ACTIONS**

NONE

WINDOW A06
(Continued)

3.0 INITIAL OPERATOR ACTIONS

> NOTE
>
> This procedure should be continued even if the reactor is tripped.

1. IF the reactor is NOT tripped AND within one hour of receiving the alarm, **check** the PS1 and PS2 "Power On" status lights on the outside of listed cabinets located in the Control Building in room RB 71.

1-1606-U3-FLR-001	Power Cabinet 1BD
1-1606-U3-FLR-002	Power Cabinet 1AC
1-1606-U3-FLR-003	Power Cabinet 2BD
1-1606-U3-FLR-004	Power Cabinet 2AC
1-1606-U3-FLR-005	Power Cabinet SCDE

a. **Perform** 13502-1, Section 4.4.6 to test backup batteries for PS1 and PS2 in each power cabinet.

b. IF both PS1 and PS2 power supply lights are lit AND backup batteries test good for all Power Cabinets, **Go To** step 3.6.

c. IF only one power supply light (either PS1 or PS2) for a Power Cabinet is lit, perform the following:

(1) **Reset** the respective power supply:

Open the cabinet door and **locate** switches S6 and S7 in the lower left-hand area of the monitoring test cabinet.

Depress switch S6 to reset PS1 or S7 to reset PS2. **Check** the status light for PS1 or PS2 ON.

(2) IF the power supply will NOT reset, **pull** and **caution tag** the 120V AC supply fuses to the failed power supply and timer to prevent possible damage to the Reset Timer relays:

Power Supply PS1	FU1A and FU1B
Power Supply PS2	FU2A and FU2B

(3) **Go to** step 3.6.

WINDOW A06
(Continued)

d. IF neither PS1 and PS2 power supply lights are lit, attempt to **reset** at least one power supply by opening the cabinet door and locate switches S6 and S7 in the lower left hand area of the monitoring test cabinet. **Depress** switch S6 to reset PS1 or S7 to reset PS2; **Check** the status light for PS1 or PS2 ON.

NOTE

Communications should be established between the control room and the rod control cabinets prior to continuing with this procedure.

CAUTION

IF PS1 or PS2 are not reset, power to the holding coils is supplied from the back-up batteries. IF PS1 or PS2 are not reset within one hour, rod drops may occur due to low voltage on the holding coil.

2. IF no power supply will reset, **notify** I&C to immediately initiate action to restore at least one power supply in each power cabinet and to monitor voltage at test point E1 to neutral.

3. IF voltage at test point E1 has degraded to less than 19.5V DC or rods drop, **trip** the reactor and **Go To** 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION, while continuing actions in this procedure.

4. For any power supply that will NOT reset, **pull** and **caution tag** the 120V AC supply fuses to the failed power supply and timer to prevent possible damage to the Reset Timer relays:

Power Supply PS1	FU1A and FU1B
Power Supply PS2	FU2A and FU2B

5. WHEN or IF the reactor is tripped AND within two hours of receiving the alarm, **pull** and **caution tag** the battery fuse to both power supply back-up batteries to prevent possible draining the battery.

Battery for Power Supply PS1	F5A
Battery for Power Supply PS2	F6A

WINDOW A06
(Continued)

6. **Notify** I&C to determine which power supply has failed by checking the output voltage of all Power Cabinet and Logic Cabinet power supplies and repair as needed.

7. **Check**, **replace** fuses and **remove** caution tags when power is ready to be restored to the power supplies and batteries.

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE

The Non Urgent Failure alarm will automatically reset when the malfunction is corrected.

Notify I&C personnel to investigate and correct the cause of the alarm.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AT01-573, 1X6AT01-574, 1X6AT01-575, 1X6AT01-576, 1X6AT01-605, 1X3D-BD-R01C

WINDOW B06

ORIGIN	SETPOINT	ROD CONTROL URGENT FAILURE
Power Cabinet Logic Cabinet	Not Applicable	

1.0 **PROBABLE CAUSE**

1. Power Cabinet Urgent Failure:

   a. Phase fault - voltage to coils has excessive ripple due to a blown fuse or thyristor that has lost gate control.

   b. Regulation failure - the coil current does not match current order within a preset time or full current is on too long.

   c. Multiplexing failure - power is being supplied to a movable or lift coil when movement of that rod has not been commanded.

   d. Logic failure - simultaneous zero current orders to stationary and movable grippers.

   e. Loose Card - loose or removed printed circuit card.

2. Logic Cabinet Urgent Failure:

   a. Pulser fails to generate pulses when signaled.

   b. Slave Cycler receives "Go" order signal before completing previous step.

   c. Loose circuit card.

WINDOW B06
(Continued)

2.0 AUTOMATIC ACTIONS

Prevents automatic and manual rod motion by performing the following:

POWER CABINET URGENT FAILURE

CAUTIONS

- Rods powered from the unaffected Power Cabinets can be moved in INDIVIDUAL BANK SELECT. However, IF the cause of the alarm is a loss of current to the stationary gripper (regulation failure) THEN moving the bank selector switch may cause the affected group of rods to drop.
- IF the cause of the alarm is a logic failure in the Power Cabinet, THEN resetting the alarm from the QMCB or locally may cause ratcheting of the rods.

1. Sends an inhibit signal to the PULSER/OSCILLATOR when the affected group is selected to move.
2. Supplies holding current to the movable and stationary grippers and no current to the lift coils for the affected group.
3. Sends an inhibit signal to the group step counter for the affected group.

WINDOW B06
(Continued)

LOGIC CABINET URGENT FAILURE

NOTES

- An Urgent Failure in the Logic Cabinet main circuits will prevent all rod motion for all of the control banks and shutdown banks A and B only. Shutdown banks C, D, and E will not be affected unless the alarm is caused by a loose or missing card in which case Shutdown banks C, D, and E will not be allowed to move either.
- An Urgent Failure in the Shutdown Banks C, D, and E portion of the Logic Cabinet will prevent all rod motion for Shutdown banks C, D, and E only. The control banks and shutdown banks A and B will not be affected unless the alarm is cause by a loose or missing card in which case the control banks and shutdown banks a and B will not be allowed to move either.

1. Sends an inhibit signal to the PULSER/OSCILLATOR

3.0 **INITIAL OPERATOR ACTIONS**

IF all rod motion has NOT stopped, **Go To** 18003-C, "Rod Control System Malfunction".

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NOTES

- The Rod Control Urgent Failure alarm seals in and must be reset using the Rod Control Alarm Reset Handswitch, 1-HS-40039, when the condition causing the alarm has cleared.
- Use of 1-HS-40039 resets the alarm circuits, demands full latching current and resets the MASTER CYCLER.

1. **Stabilize** Tavg, using turbine load and boration or dilution.
2. **Notify** appropriate plant personnel to investigate and correct the cause of the alarm.
3. **Refer To** 13502-1 to reset rod control components.

WINDOW B06
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AT01-573, 1X6AT01-574, 1X6AT01-575, and 1X6AT01-576

WINDOW C05

ORIGIN	SETPOINT	
Control Board Display Panel	Not Applicable	RPI NON URGENT ALARM

1.0 PROBABLE CAUSE

1. Card failure or removal.
2. Detector coil or cable failure.
3. Power supply failure.

2.0 AUTOMATIC ACTIONS

1. Data A Failure or Data B Failure LEDs flash depending on the cause and location of problem.
2. The General Warning LEDs flash for the affected rod(s).

> NOTE
>
> DRPI DATA A FAILURE results in system accuracy of +10, -4 steps. DRPI DATA B FAILURE results in system accuracy of +4, -10 steps.

3. DRPI goes to Half Accuracy Mode for the affected rod(s).

3.0 INITIAL OPERATOR ACTIONS

IF Dynamic Rod Worth measurement is in progress for Physics Testing, follow the Physics Testing Procedures Guidance for Operator actions in response to a DRPI Non-Urgent Failure Alarm.

IF Physics Testing is NOT in progress AND IF this alarm is coming in on an intermittent basis, THEN **Refer To** 13502-1, “Control Rod Drive And Position Indication System”, section for operation with DRPI at half accuracy.

WINDOW C05
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Notify** I&C personnel to investigate and correct the cause of the alarm.

2. IF it is determined that rod(s) are misaligned, **Go To** 18003-C, "Rod Control System Malfunction".

3. **Refer To** 13502-1, “Control Rod Drive And Position Indication System”, section for operation with DRPI at half accuracy.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCE: 1X6AT02-187

WINDOW D05

ORIGIN	SETPOINT	
Control Board Display Panel	Not Applicable	RPI URGENT ALARM

1.0 **PROBABLE CAUSE**

1. Simultaneous failure of DATA A and DATA B.
2. Greater than 1 BIT difference between DATA A and DATA B - DRPI sees the affected rod(s) at two different positions.
3. Sum of DATA A and DATA B is greater than 38 - DRPI sees the affected rod(s) at greater than 228 steps withdrawn.

2.0 **AUTOMATIC ACTIONS**

1. DRPI Control Board Displays:
   a. URGENT ALARM 1, 2, 3 LEDs flash.
   b. DATA A and DATA B FAILURE LEDs.
   c. GENERAL WARNING LED(s) for the affected rod(s)
   d. ROD BOTTOM LED(s) for the affected rod(s)
2. QMCB alarms
   a. ROD AT BOTTOM and/or TWO OR MORE RODS AT BOTTOM.

WINDOW D05
(Continued)

3.0 INITIAL OPERATOR ACTIONS

IF Dynamic Rod Worth measurement is in progress for Physics Testing, follow the Physics Testing Procedures' Guidance for operator actions in response to a DRPI Urgent Failure Alarm.

IF Dynamic Rod Worth measurement is NOT in progress AND IF any rod is NOT fully inserted:

MODES 1 & 2

1. Immediately **place** DRPI ACCURACY MODE switch to the A ONLY or B ONLY position.

2. IF the RPI URGENT ALARM does NOT clear in either the A ONLY or B ONLY position, **place** rods in MANUAL and minimize rod motion.

3. IF the RPI URGENT ALARM clears in either the A ONLY or B ONLY position, **operate** DRPI using the half accuracy mode per 13502-1, "Control Rod Drive and Position Indication System".

MODES 3, 4, & 5

1. Immediately **place** DRPI ACCURACY MODE switch to the A ONLY or B ONLY position.

2. IF the RPI URGENT ALARM does NOT clear when the DRPI ACCURACY MODE switch is in either the A ONLY or B ONLY position:

    a. Immediately **open** the reactor trip breakers per TR 13.1.8 or 13.1.9.

    b. **Return** the DRPI ACCURACY MODE switch to DATA A+B per 13502-1, as determined by the SS.

IF this alarm is coming in on an intermittent basis, THEN **Refer To** 13502-1, "Control Rod Drive And Position Indication System", section for operation with DRPI at half accuracy.

WINDOW D05
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Refer To** Technical Specification LCO 3.1.7 and Technical Requirements TR 13.1.8 and 13.1.9.

2. **Notify** Maintenance (I&C) to determine the actual rod position of rod(s) in question and begin recording rod positions in accordance with 14915-1, "Special Condition Surveillance Logs".

3. WHEN the DRPI problems have been resolved, **return** DRPI to normal per 13502-1.

5.0 COMPENSATORY OPERATOR ACTIONS

1. **Initiate** Data Sheet 4 of 14915-1, "Special Conditions Surveillance Log".

2. **Log** corrective actions to repair the disabled annunciator or reasons for no action on 10018-C, "Annunciator Control", Figure 2.

3. **Log** compensatory actions on 10018-C, "Annunciator Control", Figure 5.

END OF SUB-PROCEDURE

REFERENCE: 1X6AT02-187

1. 015A2.04 001/LOIT/RO/M/F 3.3/3.8/015A2.04/LO-LP-39206-06///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- A spurious turbine runback occurs and is terminated at 80% reactor power.
- ALB10-F06 DELTA FLUX DEVIATION is received.

Which one of the following completes the following statement?

Per Tech Spec 3.2.3, "Axial Flux Difference (AFD)," the AFD is considered outside limits when a **minimum** of __(1)__ Power Range NI channels indicate outside AFD limits,

and

with AFD outside of the Tech Spec 3.2.3 limits, thermal power must be reduced to __(2)__ within 30 minutes.

	__(1)__	__(2)__
A.	1	< 50%
B.	1	< 75%
C.✓	2	< 50%
D.	2	< 75%

K/A

015 **Nuclear Instrumentation**

A2.04 **Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:**

- Effects on axial flux density of control rod alignment and sequencing, xenon production and decay, and boron vs. control rod reactivity changes

K/A MATCH ANALYSIS

The question establishes a scenario where rod motion has resulted in an AFD transient.

Utilizing NI's, the candidate is required to predict when AFD will be outside Tech Spec required limits. Additionally, the candidate is required to utilize Tech Spec guidance to mitigate the consequences of the AFD transient.

EXPLANATION OF REQUIRED KNOWLEDGE

The described turbine runback will result in auto rod insertion. Depending on the magnitude of the runback, control rods may insert well below the Rod Insertion Limit. This large rod transient will affect AFD. Per the Note above the line in TS 3.2.3, AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD is outside limits.

With AFD outside limits, TS 3.2.3 Cond A states that reactor power must be reduced to <50% RTP within 30 minutes. Since this is a <1hr Tech Spec action, it is RO required knowledge. Additionally, ARP 17010-C for window F06 also directs lowering power to <50% with AFD outside limits.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per the Note above the line in TS 3.2.3, AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD is outside limits. However, most Tech Spec limits are exceeded when a single instrument is outside the required value. It is reasonable for a candidate without specific knowledge of the Note for TS 3.2.3 to apply this generic knowledge concept and believe AFD is out of spec based on a single NI channel.

The second part is correct. With AFD outside limits, TS 3.2.3 Cond A states that reactor power must be reduced to <50% RTP within 30 minutes.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. With AFD outside limits, TS 3.2.3 Cond A states that reactor power must be reduced to <50% RTP within 30 minutes. However, other specs have power reduced below 75% as required action. For example, TS 3.2.4 with one Power Range Nuclear instrument inoperable, the QPTR action would not be applicable if Reactor Power was reduced below 75%. Additionally, TS 3.1.4 would required thermal power reduced to <75% if rod alignments are not within limits and SDM cannot be verified.

C. Correct. The first part is correct. Per the Note above the line in TS 3.2.3, AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD is outside limits

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	015A2.04
Importance Rating:	3.3 / 3.8

Technical Reference: ARP 17010-C, Rev 50.0, pages 65 & 66
TS 3.1.4, Amendment No. 96, pages 3.1.4-1 & 2
TS 3.2.3, Amendment No. 158, page 3.2.3-1
TS 3.2.4, Amendment No. 96, pages 3.2.4-1 thru 4

References provided: None

Learning Objective: LO-LP-39206-06 State the action required for being outside the band at various power levels.
LO-LP-39206-01 For any item in section 3.2 of Tech Specs, be able to:
a. State the LCO.
b. State any one hour or less required actions.

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.1 / 41.5 / 41.10

Comments:

You have completed the test!

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable.	A.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.3	Verify SDM is ≥ the limit specified in the COLR.	Once per 12 hours
	AND		
	B.4	Perform SR 3.2.1.1.	72 hours
	AND		
	B.5	Perform SR 3.2.2.1.	72 hours
	AND		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

---NOTE---
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE------------- Required Action A.6 must be completed whenever Required Action A.5 is implemented. --------------------------------- QPTR not within limit.	A.1 Limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00. AND A.2.1 Perform SR 3.2.4.1. AND A.2.2 Limit THERMAL POWER to ≥ 3% below RTP for each 1% QPTR > 1.00. AND A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	2 hours Once per 12 hours -----------NOTE---------- For performances of Required Action A.2.2 the Completion Time is measured from the completion of SR 3.2.4.1. ----------------------------- 2 hours Within 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Actions A.1 and A.2.2 (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)		AND Once per 7 days thereafter
	AND	
	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
	AND	
	A.5 -------------NOTE------------- Perform Required Action A.5 only after Required Action A.4 is completed. ---------------------------------- Calibrate excore detectors to show QPTR = 1.00.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	-------------NOTE------------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------------------------------- Perform SR 3.2.1.1 and SR 3.2.2.1.	-----------NOTE---------- Only one of the following Completion Times, whichever becomes applicable first, must be met. ----------------------------- Within 24 hours after reaching RTP <u>OR</u> Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to $\leq$ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	---------------------------NOTE----------------------------- With one power range channel inoperable, the remaining three power range channels can be used for calculating QPTR. --- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2	----------------------------NOTE----------------------------- Only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER ≥ 75% RTP. --- Confirm that the normalized symmetric power distribution is consistent with QPTR.	Once within 12 hours <u>AND</u> In accordance with the Surveillance Frequency Control Program

WINDOW F06

ORIGIN	SETPOINT	DELTA FLUX DEVIATION
YC-1140	Plant Technical Data Book, Tab 6	

1.0 **PROBABLE CAUSE**

1. Xenon transient
2. Control rod motion
3. Thermal power transient
4. Loss of Power Range Detector voltage.
5. IPC Failure

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

> NOTE
>
> The Delta Flux Deviation Program satisfies the requirement for “Verify AFD within limits for each OPERABLE excore channel” in Technical Specifications SR 3.2.3.1.

NONE

WINDOW F06
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE

This annunciator comes in immediately upon exceeding the limits for acceptable operation.

1. With Reactor Power greater than 50% **check** differential flux indications and if two or more are outside the limits, perform the following:

    a. **Restore** the indicated AFD to within the limits, or

    b. **Reduce** Thermal Power to less than 50% of Rated Thermal Power within 30 minutes.

2. Thermal Power shall not be increased above 50% of Rated Thermal Power until the indicated AFD is within the limits specified by the COLR.

3. IF loss of Power Range Detector voltage is determined, **Go To** 18002-C, "Nuclear Instrumentation System Malfunction".

4. **Refer To** Technical Specification LCO 3.2.3.

5. IF alarm is inoperable, begin recording differential flux in accordance with 14915-1, "Special Condition Surveillance Logs" (Technical Specifications SR 3.2.3.1).

6. **Refer To** Plant Computer alarm summary display for additional information relating to this alarm.

5.0 COMPENSATORY OPERATOR ACTIONS

Initiate Data Sheet 6 of 14915-1, "Special Condition Surveillance Logs"

END OF SUB-PROCEDURE

REFERENCES: Technical Specification Section LCO 3.2.3

1. 015AA1.23 001/LOIT/RO/M/F 3.1/3.2/015AA1.23/LO-TA-16007///

Initial condition:

- Unit 1 is at 100% reactor power.

Current condition:

- ALB08-E03 RCP 1 VIBRATION ALERT is received.

Which one of the following completes the following statement?

Per the applicable Annunciator Response Procedure, the operators are directed to monitor the RCP vibration readings on the __(1)__,

and

per 13003-1, "Reactor Coolant Pump Operation," the RCP maximum operating limit for **frame** vibration is __(2)__ mils.

	__(1)__	__(2)__
A.	plant computer	5
B.	plant computer	20
C.✓	local vibration monitoring panel	5
D.	local vibration monitoring panel	20

K/A

015 RCP Malfunctions

A1.04 Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions:

- RCP vibration

K/A MATCH ANALYSIS

The question tests the candidates ability to monitor RCP vibration by selecting the correct monitoring location and operate (ie, shut down) the RCP as needed based on vibration readings.

EXPLANATION OF REQUIRED KNOWLEDGE

RCP vibration is monitored using Control Room annunciators ALB07-E03, E04, F03, and F04. These alarms come in at a FRAME vibration of 3 mils or a SHAFT vibration of 15 mils. ARP 17008-1 directs dispatching an operator to the local panel

1-1201-P5-VMP to determine actual vibration levels. The IPC has a screen to monitor most of the RCP support parameters; however, RCP vibration cannot be monitored in the Control Room.

Per SOP 13003-1 Limitation 2.2.10, an RCP shall be stopped if SHAFT vibration exceeds 20 mils or FRAME vibration exceeds 5 mils. ARP 17008-1 supports these limits and directs the operator to SOP 13003-1. ARP 17008-1 contains additional requirements to shutdown the RCP if the rate of vibration increase exceeds 1 mil/hr SHAFT or .2 mils/hr FRAME.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. RCP vibration readings can only be obtained locally at panel 1-1201-P5-VMP. However, all other RCP operating parameters are displayed on the IPC. A candidate without specific knowledge of where and how to obtain vibration readings could assume that the IPC would display vibration data also.

The second part is correct. Per SOP 13003-1 Limitation 2.2.10, an RCP shall be stopped if the FRAME vibration exceeds 5 mils.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per SOP 13003-1 Limitation 2.2.10, an RCP shall be stopped if the FRAME vibration exceeds 5 mils. It also states that an RCP shall be stopped if SHAFT vibration exceeds 20 mils. It is common for operators to confuse the frame and shaft limits.

C. Correct. The first part is correct. RCP vibration readings can only be obtained locally at panel 1-1201-P5-VMP.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G1	
K/A#	015AA1.23	
Importance Rating:	3.1 / 3.2	
Technical Reference:	SOP 13003-1, Rev 47.1, page 7 ARP 17008-1, Rev 18.0, pages 36 thru 38	
References provided:	None	
Learning Objective:	LO-PP-16401, Rev 5.4, slides 30 & 31	
	LO-TA-16007	Obtain RCP vibration data in response to RCP vibration alarms per ARP 17008-1
	LO-TA-16001	Start a RCP using 13003-1
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.7 / 41.10	
Comments:		

You have completed the test!

INITIALS

2.2.8 The following starting duty cycle for the RCP should be observed: ________

- Only one RCP shall be started at any one time.
- Two successive starts are permitted, provided the motor is permitted to coast to a stop between starts.
- A third start may be made when the winding and core have cooled by running for a period of 20 minutes, or by standing idle for a period of 45 minutes.

2.2.9 During RCS filling and venting, RCS pressure must be greater than 325 psig prior to starting an RCP to verify adequate seal D/P is maintained throughout RCS fill and vent. If necessary, the RCP should be stopped prior to seal D/P dropping less than 200 psid. If the seal D/P goes below 200 psid during pump operation or coast down, the RCP should be evaluated before restarting the RCP. ________

2.2.10 An RCP shall be stopped IF any of the following conditions exist: ________

- Motor bearing temperature exceeds 195°F.
- Motor stator winding temperature exceeds 311°F.
- Seal water inlet temperature exceeds 230°F
- Total loss of ACCW for a duration of 10 minutes.
- RCP shaft vibration of 20 mils or greater.
- RCP frame vibration of 5 mils or greater.
- Differential pressure across the number 1 seal of less than 200 psid.

2.2.11 If a loss of RCP seal cooling (Seal Injection and/or ACCW to Thermal barrier) occurs, resulting in RCP shutdown due to exceeding operating limits, then the unit should be cooled down to Mode 5 to facilitate recovery. Upon reaching Mode 5, ACCW to the Thermal barrier should be restored. Seal injection should then be returned to service. This sequence should prevent seal damage, RCP shaft bowing, ACCW System damage, etc. due to excessive thermal stresses. ________

WINDOW E03

ORIGIN	SETPOINT
1-XE-0471A,B	3 MILS FRAME
1-XE-0471C,D	15 MILS SHAFT

RCP 1
VIBRATION
ALERT

1.0 **PROBABLE CAUSE**

1. Pump Bearing failure.
2. Pump Impeller - shaft assembly out-of-balance.
3. Misalignment between Pump Shaft and Motor Shaft.
4. RCS operating temperature below 500°F.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW E03
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

> NOTE
>
> The Vibration Monitoring Panel displays auctioneered high vibration levels.

1. **Dispatch** an operator to the Vibration Monitoring Panel 1-1201-P5-VMP to:

   a. **Check** both vibration channels and alarm setpoints for shaft and frame of RCP 1 (4 points in all) to verify no obvious vibration monitoring equipment problems exist.

   b. **Notify** maintenance to verify alarm condition.

   c. **Log** any RCP Vibration LEDs illuminated and any elevated vibration readings in Control Room Electronic Log.

> NOTE
>
> If alarming condition has cleared, holding master reset in the depressed position for 2-3 seconds will clear all alarms for all RCPs.

> **CAUTION**
>
> IF alarming condition has not cleared, system engineer should be contacted prior to approving resetting any alarms.

   d. When SS directs, attempt to **reset** alarm by pressing the Black Master Reset Button located on left side of bottom Card Panel above key switch.

   e. A condition report should be written to capture this event.

2. **Continue operation** of affected RCP 1 and frequently monitor vibration.

3. **Refer** to 13003-1, "Reactor Coolant Pump Operation" and **shut down** RCP 1 if rate of frame vibration increase exceeds .2 MILS/hour.

4. **Refer** to 13003-1, "Reactor Coolant Pump Operation" and **shut down** RCP 1 if rate of shaft vibration increase exceeds 1 MIL/hour.

WINDOW E03
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB113, 1X6AB09-119, 1X3D-BD-M01A, 1X3D-CD-M10A, 1X6AB09-88, CX5DT101-176A, CX5DT101-176B

1. 022AK3.01 001/LOIT/RO/C/A 2.7/3.1/022AK3.01/LO-TA-16009/LO-PP-09///

Initial condition:

- Unit 1 is at 100% reactor power.

Current condition:

- CVCS makeup capability is lost.

Which one of the following completes the following statement?

As VCT level begins to slowly lower, VCT pressure is __(1)__ maintained.

If VCT pressure were to lower from 25 to 18 psig, RCP seal #1 leak-off flow rates would __(2)__.

	__(1)__	__(2)__
A.	automatically	decrease
B.✔	automatically	increase
C.	manually	decrease
D.	manually	increase

K/A

022 Loss of Reactor Coolant Makeup

AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:

- Adjustment of RCP seal backpressure regulator valve to obtain normal flow

K/A MATCH ANALYSIS

The question examines the candidates knowledge of the operation of the VCT Hydrogen pressure regulator. VCT pressure establishes the backpressure on RCP seal leakoff. The question addresses the type of regulator (manual or automatic) and the reason for maintaining constant pressure.

EXPLANATION OF REQUIRED KNOWLEDGE

At power, VCT Hydrogen pressure is established by Chemistry to regulate RCS Hydrogen concentration. The allowable pressure band is 18-45 psig. The lower limit is bounded by RCP seal #1 leakoff and the upper limit by boric acid emergency boration flow. Any change in VCT pressure affects RCP seal #1 leakoff, which in turn affects

RCP seal #2 flow.

As VCT level lowers, the hydrogen regulator valve will open and automatically bring hydrogen pressure back to setpoint. As VCT level increases, hydrogen pressure increases and more gas flows into the waste gas system, which eventually brings pressure back down. If VCT level changes quickly, a change in hydrogen pressure will be observed because the control system response lags. If VCT level is changed slowly, hydrogen pressure will remain essentially constant.

VCT pressure creates a back pressure for RCP seal #1 leakoff. This backpressure is required to force water from the #1 seal up and through the #2 and #3 seals. As VCT pressure lowers, seal #1 leakoff increases and the flow to seal #2 decreases.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. As VCT level decreases, the hydrogen regulator valve will open and automatically bring hydrogen pressure back to setpoint.

The second part is incorrect and assumes the candidate does not understand the system configuration and relationship between RCP seal leakoff and the VCT. If the candidate does not know that VCT pressure provides backpressure to seal leakoff, then the candidate may find it reasonable to assume that as VCT pressure lowers, flow would also lower. In addition, the candidate may confuse the #1 and #2 seal response, and assume that as VCT pressure lowers, #1 seal flow decreases.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. As VCT pressure lowers, back pressure on the #1 seal leakoff also lowers, resulting in an increase in seal #1 leakoff flow.

C. Incorrect. Plausible. The first part is incorrect. PCV-8156 is a regulator valve which controls VCT hydrogren pressure. However, a candidate unfamiliar with hydrogen makeup to the VCT could reasonably assume this regulator is a manual valve, since raising VCT pressure during normal operation requires manual adjustment of the regulator.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	022AK3.01
Importance Rating:	2.7 / 3.1

Technical Reference: ARP 17007-1 Rev 29.1
SOP 13007-1, Rev 34.5, page 7
V-LO-PP-16401, Rev 5.4, slides 12 thru 14

References provided: None

Learning Objective: LO-PP-09200-12 Describe how the following affects seal injection and seal return flow:
a. RCS pressure changes
b. VCT pressure changes
c. Safe injection signal

LO-TA-16009 Respond to abnormal RCP seal per 13003-1/2

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 41.10 / 45.6 / 45.13

Comments:

You have completed the test!

INITIALS

4.0 INSTRUCTIONS

4.1 ALIGNING VCT HYDROGEN PURGE - NORMAL OPERATION

4.1.1 **Request** Chemistry to verify, by sample analysis, the nitrogen content in the VCT gas space. ________

4.1.2 IF a nitrogen atmosphere exists in the VCT, **align** the tank for hydrogen purge operation per Section 4.3. ________

4.1.3 **Request** Chemistry to verify, by sample analysis, that the oxygen concentration in the VCT gas space is less than 2% by volume. ________

4.1.4 IF the VCT oxygen concentration limit is approached, **lower** the oxygen content using Section 4.4. ________

4.1.5 **Verify** a hydrogen atmosphere exists in the VCT as follows:

a. Open 1-2406-U4-001 HYDROGEN SUPPLY TO CVCS VCT ISOLATION AB-A24. ________

b. Open 1-1208-U4-107 VCT Hydrogen Manifold Isolation, AB A24. ________

c. Close 1-1208-U4-108 VCT Nitrogen Manifold Isolation, AB A24. ________

d. Close 1-1208-U4-352 Waste Gas Decay Shutdown Tank Supply To VCT, AB A47. ________

4.1.6 **Check** that VCT Hydrogen Regulator 1-PCV-8156 is set to maintain 18 psig or greater AB PIPE CHASE ROOM A24. ________

4.1.7 IF the Hydrogen Regulator requires adjustment, **loosen** set screw and **adjust** point to raise or lower pressure to maintain 18 psig or greater. ________

4.1.8 **Verify** the Gaseous Waste Processing System in operation, AND **aligned** to a Normal Gas Decay Tank, per 13201-1, "Gaseous Waste Processing System." ________

4.1.9 **Verify** that the VCT Purge Flow Controller 1-HIC-1094 (1-PGPP), is set at zero. ________

WINDOW F05

ORIGIN	SETPOINT	
1-PT-0115	Hi: 45 psig Low: 18 psig	VCT HI/LO PRESS

1.0 **PROBABLE CAUSE**

1. High pressure:
   a. High Volume Control Tank (VCT) level,
   b. Hydrogen or Nitrogen Pressure Regulator malfunction. (1-PCV-8156 or 1-PCV-8155).
2. Low pressure:
   a. Hydrogen or Nitrogen Pressure Regulator malfunction (1-PCV-8156 or 1-PCV-8155),
   b. Open or leaking Vent Valve,
   c. System leak.

NOTE: Low VCT level is not listed as a symptom for the low pressure condition. VCT pressure is expected to be maintained automatically on a slow event like lowering level.

2.0 **AUTOMATIC ACTIONS**

On low VCT pressure, 1-PV-0115 closes.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW F05
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Monitor** VCT pressure and level using 1-PI-0115 and 1-LI-0185.

2. IF VCT level is high:

   a. **Divert** letdown flow to the Recycle Holdup Tank (HUT position) using 1-HS-0112A on the QMCB,

   b. **Operate** makeup per 13009-1, "CVCS Reactor Makeup Control System."

   c. WHEN desired pressure/level is obtained, **place** 1-HS-0112A to the AUTO position

3. IF VCT level is normal, **adjust** VCT pressure per 13007-1, "VCT Gas Control And RCS Chemical Addition."

4. IF a system leak is suspected, **dispatch** personnel to locate and isolate the leak.

5. **Return** VCT pressure to normal as soon as possible.

6. VCT pressure should be maintained below 45 psig for the BAT pumps to be relied upon as a boration flowpath.

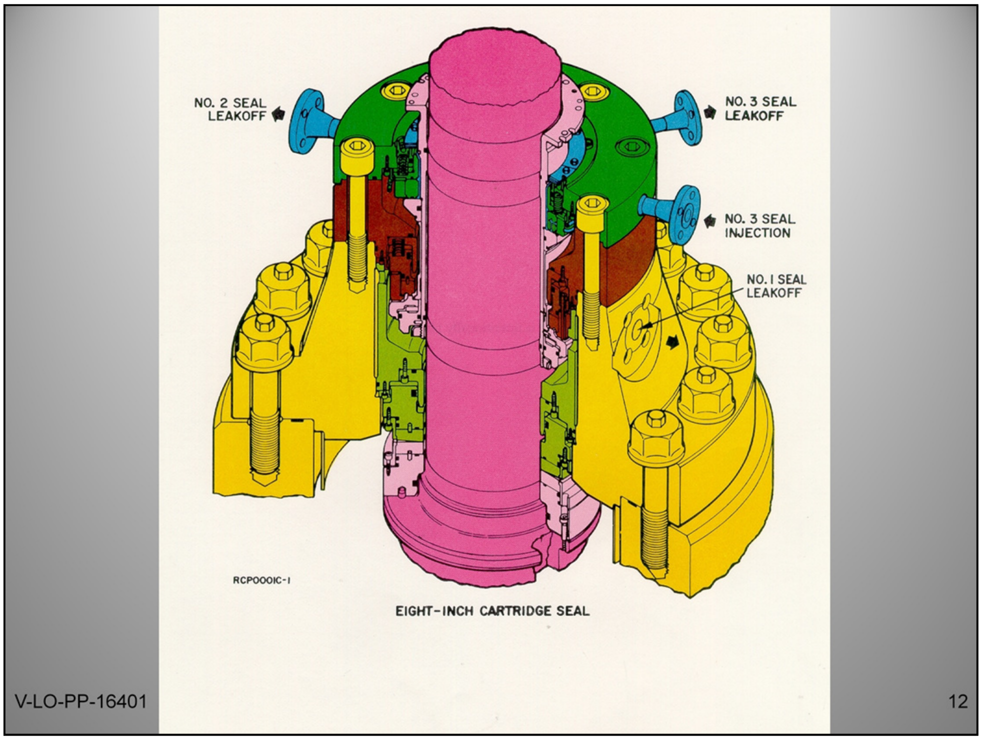
7. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB116-1, 1X4DB128, PLS

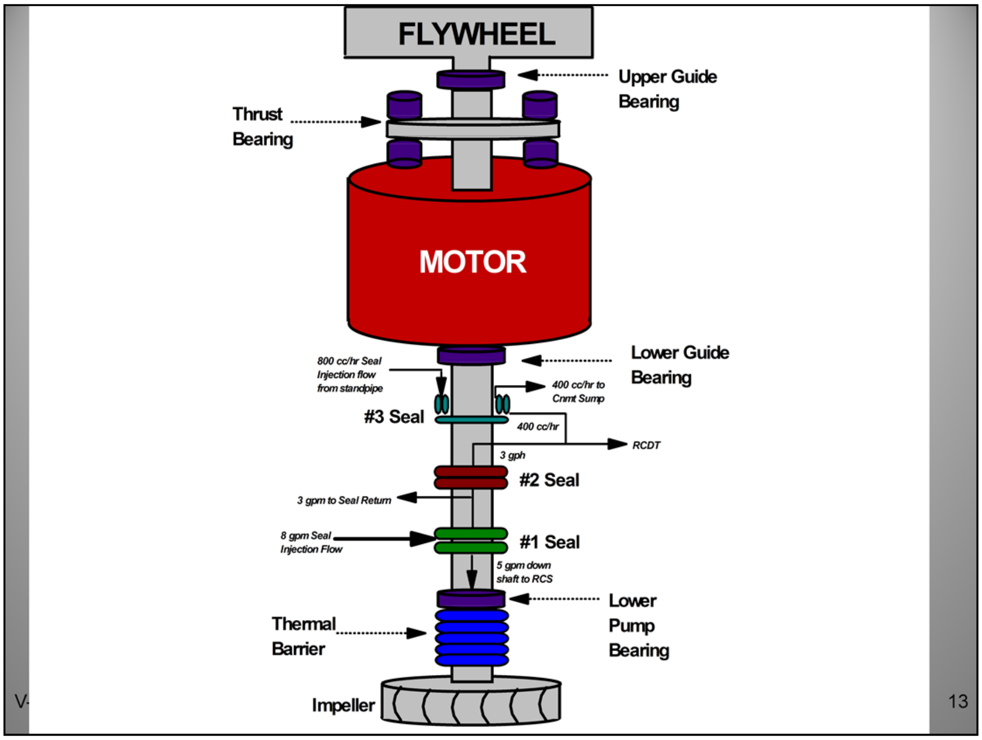


Objective 4

Demonstrate using the seal package model.

The seal package consist of three seals

1) **Number 1 seal (film riding)**
   - a) The primary seal
   - b) Seal is accomplished with a hydrostatic film between the shaft runner and seal ring.
   - c) No mechanical contact between seal ring and shaft runner (must keep ΔP >200 psid)
2) **Number 2 seal (face rubbing)**
   - a) Provides back up for #1 seal
   - b) Consist of carbon graphite (face rubbing seal)
   - c) Graphite makes contact with runner which rotates with shaft
   - d) If #1 seal fails , #2 seal converts to a film riding seal if #1 seal leak off valve is closed and seal is exposed to full RCS pressure. #2 seal designed to allow plant shutdown and should last approximately 24 hours.
   - e) Placing #2 seal in service with the RCP shaft still rotating will tend to score the shaft at the #2 seal area. This can require extensive repairs before placing the RCP back in service. Vogtle chooses to remove RCP from service and allow its shaft to come to a standstill before closing the #1 seal leak off valve to avoid this problem.
3) **Number 3 seal (face rubbing)**
   - a) Prevents the leakage of liquid and gases from the RCS into containment.
   - b) Consist of carbon graphite seal which makes contact with runner (face rubbing)
   - c) The runner is around the shaft and rotates with it.
   - d) The seal is actually two graphite sealing surfaces called dams.



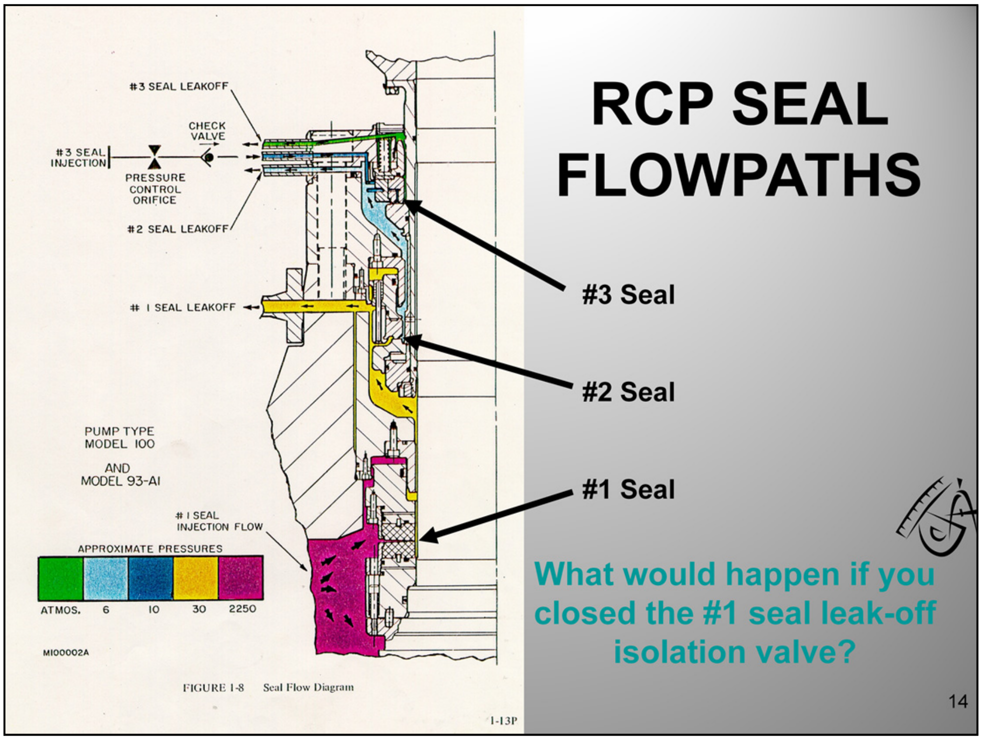
Objective 1d

2) RCP Motor Auxiliaries
- A) Motor Cooler
  - 1) Containment Air is drawn into the motor by fan blades on motor's rotor
  - 2) It is then exhausted through the motor cooler
  - 3) ACCW is the cooling medium used in the cooler (cools the outgoing air)
  - 4) This arrangement limits containment air temperature rise and in turn limits motor temperature.

3) **Flywheel**
- A) Addressed in tech spec administrative section
- B) Stores rotational energy of the pump and motor while running then releases energy by maintaining pump motion to slowly reduce core flow following loss of power for core protection.

4) RCP motor space heater
- A) Each RCP motor has a electric resistance heater.
- B) Used to prevent moisture accumulation in windings when motor is shutdown.
- C) Not needed when motor is in operation because of heat generation from motor windings.
- D) Heaters are automatically energized when either of the RCP motor breakers are opened.
- E) Heaters are supplied from 480 V MCCs

RCP motor burned up at Vogtle that was attributed to moisture from space heater breaker being open when pump was shutdown. The space heater did not energize therefore moisture accumulated while the pump was shutdown during the outage. Upon restart the motor windings shorted out. Motor rebuild was required.



Objective 2

RCP Seal Injection

a) Provided from CVCS

b) 8 gpm per pump

c) 5 gpm is directed through to lower pump radial bearing and into the RCS loop.

d) The remaining 3 gpm supplies #1 and #2 seals

e) #3 seal injection is from small tanks called Standpipes. (Gravity Fed)

f) Flow path

1) 8 gpm from CVCS enters RCP at 2250 psig
2) 5 gpm passes through the lower pump bearing lubricating and cooling it.
3) Seal injection at 2250 psig prevents RCS water from escaping the loop.
4) 3 gpm is directed through #1 seal
5) A pressure drop at 2220 psid across the #1 seal occurs.
6) Approximately 3 gph (0.05 gpm) leak off from #1 seal is used as seal injection to #2 seal.
7) The remainder of #1 seal leak off is directed back to the VCT via seal water return.
8) 3 gph passes through #2 seal and the leak off is directed to the RCDT (~5-6 psig)
9) 800 cc/hr seal injection for #3 seal is provided by standpipe (~10 psig)
10) The standpipes are located at a higher elevation than the RCP and gravity feeds #3 seal; standpipes Auto fill from RMWST.
11) #3 seal injection is injected between the two dams and sealing surfaces.
12) #3 seal injection pressure is slightly higher than #2 seal injection leak off.
13) This prevents RCS liquids or gases from escaping to the containment environment.
14) #3 seal has two leak off paths
    a) The outer dam leak off (400 cc/hr) combines with #2 seal leak off and is routed to RCDT
    b) The inner dam leak off (400 cc/hr) is directed to the containment sump.

SMART – Solid Knowledge. If the #1 seal leakoff was isolated, the #2 seal would become a film riding seal due to increased pressure across the #2 seal facing.

1. 022K3.02 001/LOIT AND LOCT/RO/C/A 3.0/3.3/022K3.02/LO-LP-36104-01///

Given the following:

- Unit 1 is in Mode 3 following a steam line break in containment.
- Only Train 'A' containment cooling units are available.
- Containment temperature is 241°F and slowly rising.
- Containment pressure is 11 psig and slowly rising.

Which one of the following completes the following statement?

SG NR level instruments on the QMCB will indicate __(1)__ than actual level,

and

the instrument inaccuracies are a **direct** result of changes in containment __(2)__.

	__(1)__	__(2)__
A.	lower	pressure
B.	lower	temperature
C.	higher	pressure
D.✓	higher	temperature

K/A

022 Containment Cooling System (CCS)

K3.02 Knowledge of the effect that a loss or malfunction of the CCS will have on the following:

- Containment instrumentation readings.

K/A MATCH ANALYSIS

The question presents the candidate with a valid scenario in which a Secondary LOCA has occurred that affects containment temperature and pressure. The candidate is required to determine the effect on SG level indication as a result of elevated containment pressure and temperature originating from the design basis accident in conjuction with loss of Containment Coolers.

EXPLANATION OF REQUIRED KNOWLEDGE

SG level transmitters utilize a closed reference leg with a condensing pot. As containment temperature increases, the reference leg density lowers, resulting in SG level indication reading higher than actual. EOPs require normal SG levels to be >10% NR to ensure a heat sink is maintained. Per the WOG EOP Setpoint Documents, >19% level must be added to compensate for impacts stemming from containment temperature under ADVERSE condtions. (Reference attached Westinghouse letter WWA5247.) Since the level transmitters utilize a closed reference leg, containment pressure has no direct impact on level indication.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. As containment temperature increases, the reference leg density lowers, resulting in SG level indication reading higher than actual. However, a candidate may either invert the density effect in the reference and variable legs or reverse the dP impacts between the two.

The second part is incorrect. Since the level transmitters utilize a closed reference leg, containment pressure has no direct impact on level indication. However, if the candidate believes SG level instruments use an open reference leg, this answer would be correct.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. As containment temperature increases, the reference leg density lowers, resulting in SG level indication reading higher than actual.

C. Incorrect. Plausible. The first part is correct. As containment temperature increases, the reference leg density lowers, resulting in SG level indication reading higher than actual.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	022K3.02
Importance Rating:	3.0 / 3.3

Technical Reference: WOG Background Documents for Adverse Containment, Westinghouse letter WWA5247

References provided: None

Learning Objective: LO-PP-29101-08 Describe routine actions taken to adjust Containment pressure and temperature.
LO-PP-29101-09 State the likely sources of Containment pressure increase during normal operations.
LO-LP-36104-01 List and describe four adverse environmental conditions that affect the reliability of instrumentation associated with critical plant parameters.

Question origin: BANK - San Onofre 2006 NRC Question # 13 022K3.02.

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 41.14

Comments:

You have completed the test!

SG NARROW RANGE LEVEL (continued)

Setpoint ID: M.2

Setpoint Value: 10%

Associated System/Component: MAIN STEAM

Applicability: COMMON

Revision: 0

Short Description:

Steam generator level just in range plus normal uncertainties

Description:

Value showing S/G level just in the narrow range including allowances for normal channel accuracy and reference leg process errors (Ref.2).

References:

1. 19000/18, 10, 26, 29, FO
   19002/FO
   19004/FO
   19010/3, 6, FO
   19012/6, FO
   19014/FO
   19031/4, 6, FO
   19100/13, 17
   19102/8
   19131/9, 34, FO
   19133/5, 7, 28, FO
   19211/7
   19222/9
   19231/9, 20
   19235/1, 5
   19242/1
   19001/1, 6, 12, FO
   19003/FO
   19005/FO
   19011/19, FO
   19013/FO
   19030/4, 7, 20, FO
   19033/4, 9, FO
   19101/7
   19121/2
   19132/4, 28, FO
   19200/F.O.3
   19221/9
   19231/7
   19233/4
   19241/1

   (jmw 4/16/93)

2. ERG Footnote Basis Document, Footnote M.02, March 2001
3. Westinghouse letter NS-OPLS-OPL-I-90-309, page 10, May 18, 1990

Key Assumptions:

1. Normal channel accuracy = 3.8%
2. Reference leg errors = 0.9%

Basis:

see INSERT M.2

<u>INSERT M.2</u>

Basis:

Narrow range level reading at the top of the U-tubes = 4.7% (Ref. 3)

Normal channel accuracy = 3.8% (Assumption 1)

Reference leg errors = 0.9% (Assumption 2)

Calculated value = 4.7% + 3.8% + 0.9%

Calculated value = 9.4%

This setpoint is used to ensure that the SG level remains above the top of the U-tubes (i.e., the U-tubes are covered). For Vogtle, the top of the U-tubes is at an elevation equivalent to 4.7% on the narrow range SG level span. This is due to the relocation of the SG narrow range level lower tap. To ensure that the U-tubes are covered, the SG narrow range level channel accuracy is added to the 4.7% of span corresponding to the elevation of the U-tubes.

The calculated value is rounded upward to the nearest ½ division (1%) to obtain the EOP setpoint value (Ref. 2).

Setpoint value = 10%

As compared to ADVERSE numbers on the following sheets.

EOP Setpoints

SG NARROW RANGE LEVEL (continued)

Setpoint ID: M.3

Setpoint Value: 32%

Associated System/Component: MAIN STEAM

Applicability: COMMON

Revision: 0

Short Description:

Steam generator level just in range plus adverse uncertainties

Description:

Value showing S/G level just in the narrow range including allowance for normal channel accuracy, post-accident transmitter errors, and reference leg process errors, not to exceed 50%. (For Unit 2)

References:

1. 19000/18, 10, 26, 29, FO
19002/FO
19004/FO
19010/3, 6, FO
19012/6, FO
19014/FO
19031/4, 6, FO
19100/13, 17
19102/8
19131/9, 34, FO
19133/5, 7, 28, FO
19211/7
19222/9
19231/9, 20
19235/1, 5
19242/1
19001/1, 6, 12, FO
19003/FO
19005/FO
19011/19, FO
19013/FO
19030/4, 7, 20, FO
19033/4, 9, FO
19101/7
19121/2
19132/4, 28, FO
19200/F.O.3
19221/9
19231/7
19233/4
19241/1

(jmw 4/16/93) 2. ERG Footnote Basis Document, Footnote M.03, March 2001

Key Assumptions:

1. Normal channel and post-accident error = 19%
2. Reference leg errors = 8%

Basis:

See INSERT M.3

INSERT M.3

Basis:

Narrow range level reading at the top of the U-tubes = 4.7% (Ref. 3)

Normal channel and post-accident error = 19% (Assumption 1)

Reference leg errors = 8% (Assumption 2)

Calculated value = 4.7% + 19% + 8%

Calculated value = 31.7%

This setpoint is used to ensure that the SG level remains above the top of the U-tubes (i.e., the U-tubes are covered). For Vogtle, the top of the U-tubes is at an elevation equivalent to 4.7% on the narrow range SG level span. This is due to the relocation of the SG narrow range level lower tap. To ensure that the U-tubes are covered, the SG narrow range level channel accuracy is added to the 4.7% of span corresponding to the elevation of the U-tubes.

The calculated value is rounded upward to the nearest ½ division (1%) to obtain the EOP setpoint value (Ref. 2).

Setpoint value = 32%

1. 024AG2.2.22 001/LOIT/RO/M/F 4.0/4.7/024AG2.2.22/LO-TA-09029///

Given the following:

- Unit 1 requires Emergency Boration.

- Shift Supervisor directs Emergency Boration through the BIT flow path.

Which one of the following completes the following statement?

For the selected Emergency Boration flow path, 13009-1, "CVCS Reactor Makeup Control System," directs the operator to establish a **minimum** flow rate to the RCS greater than __(1)__ gpm,

and

the boron concentration for the **source** used above is required to be between __(2)__ ppm.

	__(1)__	__(2)__
A.	30	2400 - 2600
B.	30	7000 - 7700
C.✓	87.5	2400 - 2600
D.	87.5	7000 - 7700

K/A

024 **Emergency Boration**

G2.2.22 **Knowledge of limiting conditions for operations and safety limits.**

K/A MATCH ANALYSIS

The question sets up a valid scenario in which an emergency boration is directed by the Shift Supervisor, requiring the candidate to determine the suction source, boron concentration, and required flow rate.

EXPLANATION OF REQUIRED KNOWLEDGE

The Shift Supervisor directs Emergency Boration through the BIT flow path. This boration will be performed per SOP 13009-1 section 4.9.4. As such, the RWST will be aligned as the boration source and a minimum flow rate of 87.5 gpm is required. Per TS SR 3.5.4.3, RWST boron concentration must be between 2400 and 2600 ppm.

Conversely, if the SS directs boration through HV-8104 or FV-0110, the suction would

be aligned to the BAST. Per SOP 13009-1, a flow rate of 30 gpm is required. Per TRM TRS 13.1.7.3, BAST concentration must be between 7,000 and 7,700 ppm.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Emergency boration through the BIT will be made with the suction aligned to the RWST and a flow rate of 87.5 gpm is required. However, if the candidate is unfamilar with the emergency boration flow paths and believes the CCP suctions will be aligned to the BAST, 30 gpm is be the correct answer.

The second part is correct. Per TS SR 3.5.4.3, RWST boron concentration must be between 2400 and 2600 ppm.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per TS SR 3.5.4.3, RWST boron concentration must be between 2400 and 2600 ppm. However, if the candidate believes boration is occuring from the BAST, a boron concentration of 7,000 - 7,700 ppm would be correct per TRM TRS 13.1.7.3.

C. Correct. The first part is correct. Emergency boration through the BIT will be made with the suction aligned to the RWST, and a flow rate of 87.5 gpm is required.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G2	
K/A#	024G2.2.22	
Importance Rating:	4.0 / 4.7	
Technical Reference:	SOP 13009-1, Rev 49.0, pages 47 & 69 TS SR 3.5.4.3, Amendment No. 158, page 3.5.4-2 TRM TRS 13.1.7.3, Rev 1-10/22/98, page 13.1-12	
References provided:	None	
Learning Objective:	LO-PP-60327-10	List the methods (flow paths/sources) available to Emergency borate the RCS.
	LO-LP-39209-01	For any given item in section 3.5 of Tech Specs, be able to: a. State the LCO. b. State any one hour or less required actions.
	LO-PP-09300-13	State the Technical Requirement, applicability, and any one hour or less actions for each of the following: f. TR 13.1.7, Borated Water Sources - Operating
	LO-TA-09029	Perform Emergency Boration using 13009-1/2
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.5 / 43.2 /45.2	
Comments:		

You have completed the test!

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-------------------------NOTE------------------------------ Only required to be performed when ambient air temperature is < 40°F. --- Verify RWST borated water temperature is ≥ 44°F and ≤ 116°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 686,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.4	Verify each sludge mixing pump isolation valve automatically closes on an actual or simulated RWST Low-Level signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. RWST inoperable.	D.1 Enter applicable Conditions of RWST Technical Specification 3.5.4.	Immediately

TECHNICAL REQUIREMENT SURVEILLANCES

---NOTES---

1. TRS 13.1.7.1 is only required to be performed when the RWST is the required borated water source.

2. TRS 13.1.7.2, TRS 13.1.7.3, and TRS 13.1.7.4 are only required to be performed when the boric acid storage tank is the required borated water source.

SURVEILLANCE		FREQUENCY
TRS 13.1.7.1	When the RWST is required to be OPERABLE, the SRs of Technical Specification 3.5.4 are applicable.	In accordance with applicable SRs.
TRS 13.1.7.2	----------------------------------NOTE------------------------------ Only required to be performed if the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is ≤ 72 °F. --- Verify the boric acid storage tank solution temperature is ≥ 65 °F (TI-0103).	7 days
TRS 13.1.7.3	Verify the boron concentration of the boric acid tank solution is between $\boxed{\text{7,000 ppm and 7,700 ppm.}}$	7 days
TRS 13.1.7.4	Verify the contained borated water volume in the boric acid storage tank is ≥ 36,674 gallons (83% of instrument span, LI-102A, LI-104A).	7 days

INITIALS

4.9.4 Emergency Boration From The RWST Through The BIT Isolation Valves

4.9.4.1 **Verify** one (1) Charging Pump is running and supplied with cooling water. ________

4.9.4.2 **Open** the following Charging Pump Suctions from the RWST:

- 1-LV-0112D ________
- 1-LV-0112E ________

4.9.4.3 **Close** the following VCT Outlet Isolations:

- 1-LV-0112B ________
- 1-LV-0112C ________

4.9.4.4 **Place** 1-LV-0112A to the HUT position. ________

4.9.4.5 **Open** the following BIT DISCH ISOLATION valves:

- 1-HV-8801A ________
- 1-HV-8801B ________

4.9.4.6 **Verify** BIT Flow (1-FI-0917A), plus total seal injection flow, minus total seal return flow is greater than 87.5 gpm. ________

4.9.4.7 **Adjust** Charging Line Flow Controller 1-FIC-0121 as necessary to maintain RCP seal injection flow at maximum flow less than 13 gpm per pump. ________

4.9.4.8 IF required for RCS inventory control, **place** an additional letdown orifice in service per 13006-1. ________

4.9.4.9 **Operate** the Pressurizer Backup Heaters as necessary to equalize boron concentrations between the RCS and the Pressurizer. ________

TABLE 1

EMERGENCY BORATION FLOW PATH ALTERNATIVES

Flow path	BATP	Valve Alignments	Other Pump Required	Flows	Flow	Note
HV8104	At least one	**OPEN** 1HV-8104	Any charging pump	>**42 GPM** 1FI-0121C	>**30 GPM** 1FI-0183A	Operate heaters
Charging Flow path	At least one	**OPEN** 1FV-0110A 1FV-0110B	Any charging pump	>**42 GPM** 1FI-0121C	>**30 GPM** 1FI-0110A	Operate heaters
RWST to Regen Hx	NA	**OPEN** 1LV-0112D 1LV-0112E **CLOSE** 1LV-0112B 1LV-0112C **HUT** 1LV-0112A	Any charging pump	>**100 GPM** 1FI-0121C	**8 to 13 GPM** seal injection flow 1HV-0182	Operate heaters
RWST to BIT	NA	**OPEN** 1LV-0112D 1LV-0112E 1HV-8801A 1HV-8801B **CLOSE** 1LV-0112B 1LV-0112C **HUT** 1LV-0112A	Any charging pump	BIT flow (1FI-0917A) + total seal flow - seal return flow >**87.5 GPM**	Adjust 1FIC-0121C to <**13 GPM** per RCP	Operate heaters
RHR (Mode 6)	NA	**OPEN** HV-8812A/B HV-8809A/B	RHR other than S/D Cooling	>100 gpm	See Proc.	Establish water removal path to prevent vessel overflow
SI (Mode 6)	NA	**OPEN** HV-8923A/B HV-8821A/B HV-8835	SI	>100 gpm	See Proc.	Establish water removal path to prevent vessel or cavity overflow

1. 026K1.01 001/LOIT/RO/M/F 4.2/4.2/026K1.01/LO-TA-13009///

Given the following:

- Unit 1 reactor trip and SI occurred due to a LOCA.
- Containment Spray (CS) actuated.

Which one of the following completes the following statement?

During the ECCS injection phase, CS pumps and ECCS pumps __(1)__ suction header(s) **penetrating** the RWST,

and

the CS pumps' sump suction valves __(2)__ automatically open on LO-LO RWST level.

	__(1)__	__(2)__
A.	share a common	will
B.✓	share a common	will NOT
C.	have separate	will
D.	have separate	will NOT

K/A

026 Containment Spray System (CSS)

K1.01 Knowledge of the physical connections and/or cause effect relationships between the CSS and the following systems:

- ECCS

K/A MATCH ANALYSIS

Question asks about the physical relationships between ECCS and the Containment Spray suction header from the RWST. Both systems share the same suction header from the RWST. The second part of the question is related to the design of the CSS when swapping suction sources and whether this is a manual or automatic swap.

EXPLANATION OF REQUIRED KNOWLEDGE

CS and ECCS pumps share a common 24" suction pipe that penetrates the RWST. This common header can be isolated by a single manual valve, 1-1204-U4-207. Once inside the Auxiliary Building, the common suction header branches off to supply the ECCS and CS pumps. Ref P&ID 1X4DB121.

ECCS containment sump suction valves automatically open on receipt of an RWST Lo-Lo Level signal following an SI actuation. The RWST semi-automatic swap over has a separate retentive circuit which allows reset of SI without resetting the RWST swap over function. During this swap over, the containment sump suction valves automatically open, and the RWST suctions remain open. The operator must close the RWST suctions with the handswitch. Ref Elementary 1X3D-BD-E03F for an example of this logic.

CS sump suction valves do not have automatic opening circuitry of any kind. These valves must be opened using the valve handswitch. Ref Elementary 1X3D-BD-J02G.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The ECCS and CS pumps share a common 24" suction header that penetrates the RWST.

The second part is incorrect. The CS sump suction valves never automatically open. These valves open only when the handswitch is taken to the open position. However, the ECCS sump suction valves do automatically open. A candidate without specific knowledge of the CS sump suction valves could find it reasonable that the CS suction valves would behave the same as the ECCS valves to prevent the CS pumps from losing suction.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. The CS sump suction valves never automatically open. These valves open only when the handswitch is taken to the open position.

C. Incorrect. Plausible. The first part is incorrect. The ECCS and CS pumps share a common 24" suction header that penetrates the RWST. However, the common suction header branches off after entering the Auxiliary Building. The branches are not common between any of the ECCS or CS pumps. A candidate without specific knowledge of ECCS and CS piping could find it reasonable that the ECCS and CS pumps have separate piping to ensure train separation. Plant walkdowns can reinforce this incorrect assumption because all the branching occurs well before the individual pump rooms.

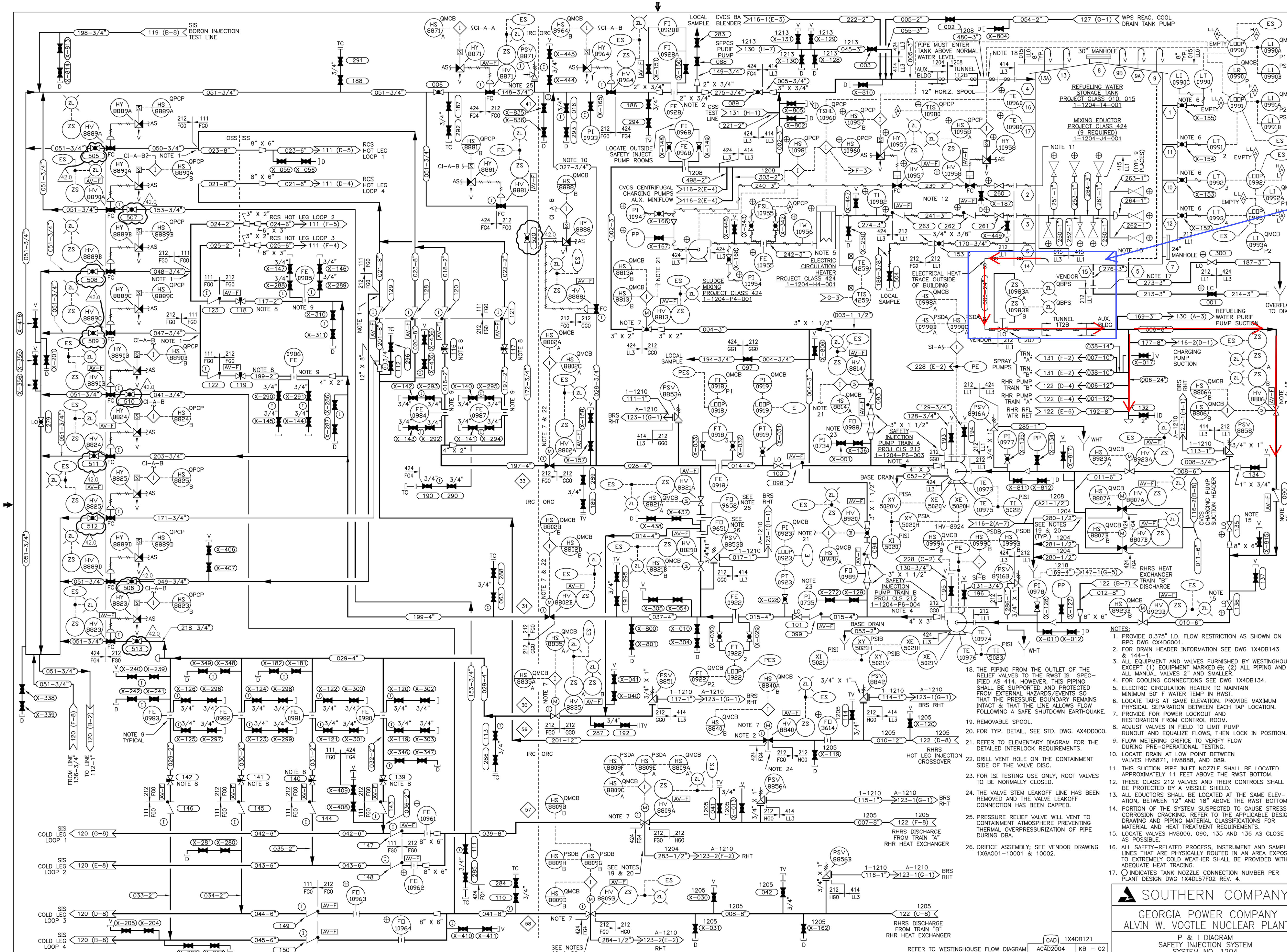
The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	026K1.01
Importance Rating:	4.2 / 4.2
Technical Reference:	P&ID 1X4DB121, Rev 42.0 Elementary 1X3D-BD-E03F, Rev 9.0 Elementary 1X3D-BD-J02G, Rev 6.0
References provided:	None
Learning Objective:	V-LO-PP-15101 Containment Spray System LO-LP-37113-02 Using EOP 19013 as a guide, briefly describe how each step is accomplished. LO-PP-15101-04 List all components that receive a Containment Spray Actuation signal and their change in status. LO-PP-15101-02 Describe what will actuate the Containment Spray System, including coincidence and set point. LO-TA-13013 Draw functional diagram of ECCS LO-TA-15005 Draw the Containment Spray System LO-TA-13009 Manually align ECCS for Cold Leg Recirculation Phase using EOP 19013-C
Question origin:	BANK - Farely 2010 NRC Question # 026K1.01
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Comments:	

You have completed the test!



NOTES:

1. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
2. FOR DRAIN HEADER INFORMATION SEE DWG 1X4DB143 & 144-1.
3. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT (1) EQUIPMENT MARKED ⊕; (2) ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
4. FOR COOLING CONNECTIONS SEE DWG 1X4DB134.
5. ELECTRIC CIRCULATION HEATER TO MAINTAIN MINIMUM 50° F WATER TEMP IN RWST.
6. LOCATE TAPS AT SAME ELEVATION. PROVIDE MAXIMUM PHYSICAL SEPARATION BETWEEN EACH TAP LOCATION.
7. PROVIDE FOR POWER LOCKOUT AND RESTORATION FROM CONTROL ROOM.
8. ADJUST VALVES IN FIELD TO LIMIT PUMP RUNOUT AND EQUALIZE FLOWS, THEN LOCK IN POSITION.
9. FLOW METERING ORIFICE TO VERIFY FLOW DURING PRE-OPERATIONAL TESTING.
10. LOCATE DRAIN AT LOW POINT BETWEEN VALVES HV8871, HV8888, AND 089.
11. THIS SUCTION PIPE INLET NOZZLE SHALL BE LOCATED APPROXIMATELY 11 FEET ABOVE THE RWST BOTTOM.
12. THESE CLASS 212 VALVES AND THEIR CONTROLS SHALL BE PROTECTED BY A MISSILE SHIELD.
13. ALL EDUCTORS SHALL BE LOCATED AT THE SAME ELEVATION, BETWEEN 12" AND 18" ABOVE THE RWST BOTTOM.
14. PORTION OF THE SYSTEM SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
15. LOCATE VALVES HV8806, 090, 135 AND 136 AS CLOSE AS POSSIBLE.
16. ALL SAFETY-RELATED PROCESS, INSTRUMENT AND SAMPLING LINES THAT ARE PHYSICALLY ROUTED IN AN AREA EXPOSED TO EXTREMELY COLD WEATHER SHALL BE PROVIDED WITH ADEQUATE HEAT TRACING.
17. ⬡ INDICATES TANK NOZZLE CONNECTION NUMBER PER PLANT DESIGN DWG 1X4DL57F02 REV. 4.
18. THE PIPING FROM THE OUTLET OF THE RELIEF VALVES TO THE RWST IS SPECIFIED AS 414. HOWEVER, THIS PIPING SHALL BE SUPPORTED AND PROTECTED FROM EXTERNAL HAZARDS/EVENTS SO THAT THE PRESSURE BOUNDARY REMAINS INTACT & THAT THE LINE ALLOWS FLOW FOLLOWING A SAFE SHUTDOWN EARTHQUAKE.
19. REMOVABLE SPOOL.
20. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
21. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK REQUIREMENTS.
22. DRILL VENT HOLE ON THE CONTAINMENT SIDE OF THE VALVE DISC.
23. FOR ISI TESTING USE ONLY, ROOT VALVES TO BE NORMALLY CLOSED.
24. THE VALVE STEM LEAKOFF LINE HAS BEEN REMOVED AND THE VALVE LEAKOFF CONNECTION HAS BEEN CAPPED.
25. PRESSURE RELIEF VALVE WILL VENT TO CONTAINMENT ATMOSPHERE PREVENTING THERMAL OVERPRESSURIZATION OF PIPE DURING DBA.
26. ORIFICE ASSEMBLY; SEE VENDOR DRAWING 1X6AG01-10001 & 10002.

REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E71 SHEET 3 OF 3.

CAD 1X4DB121 | ACAD2004 | KB - 02

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

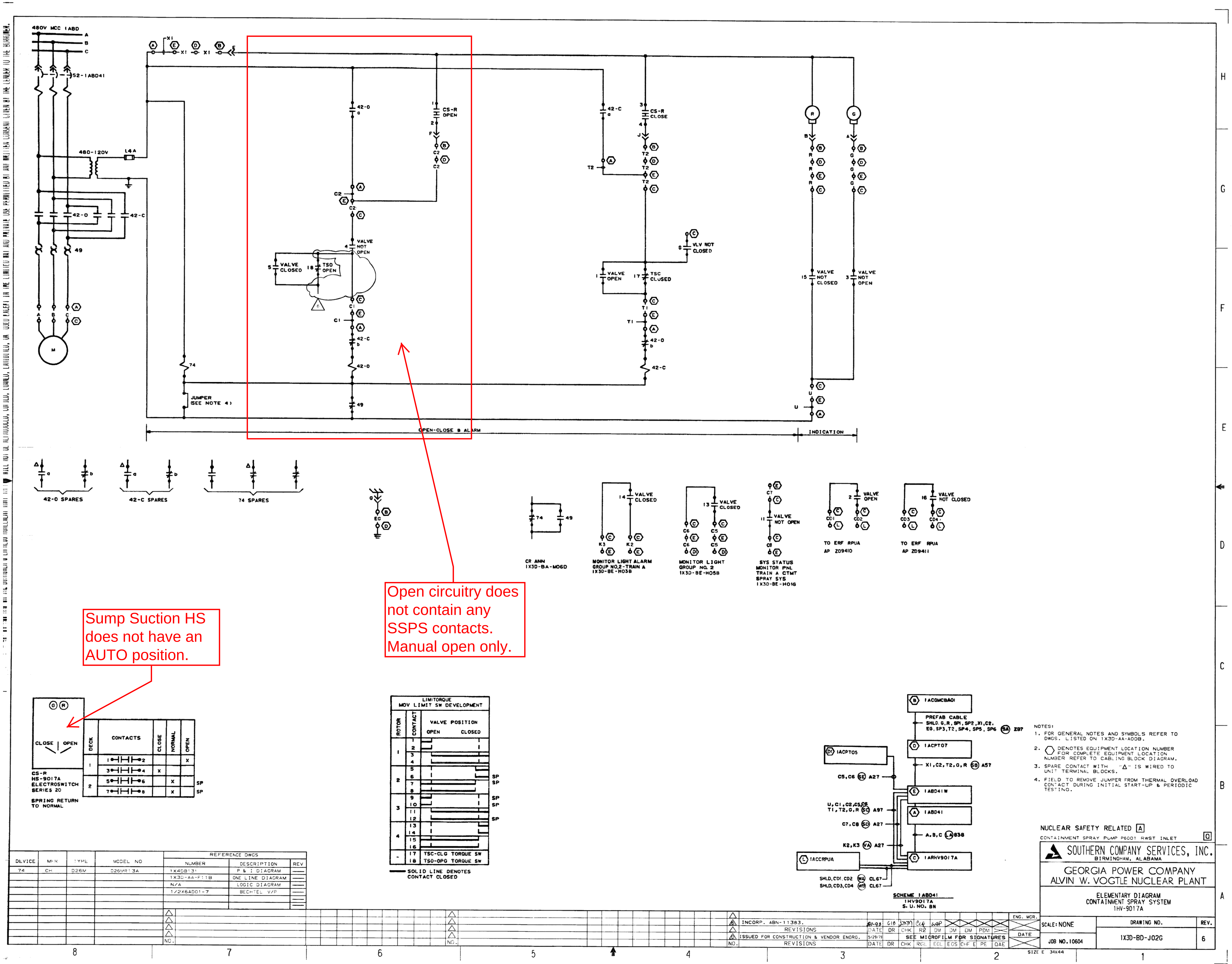
P & I DIAGRAM
SAFETY INJECTION SYSTEM
SYSTEM NO. 1204

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
42.0	REVISED PER ABN-1071073501M004, VER. 1.0	10/20/09	KB	CFC	JMR	X

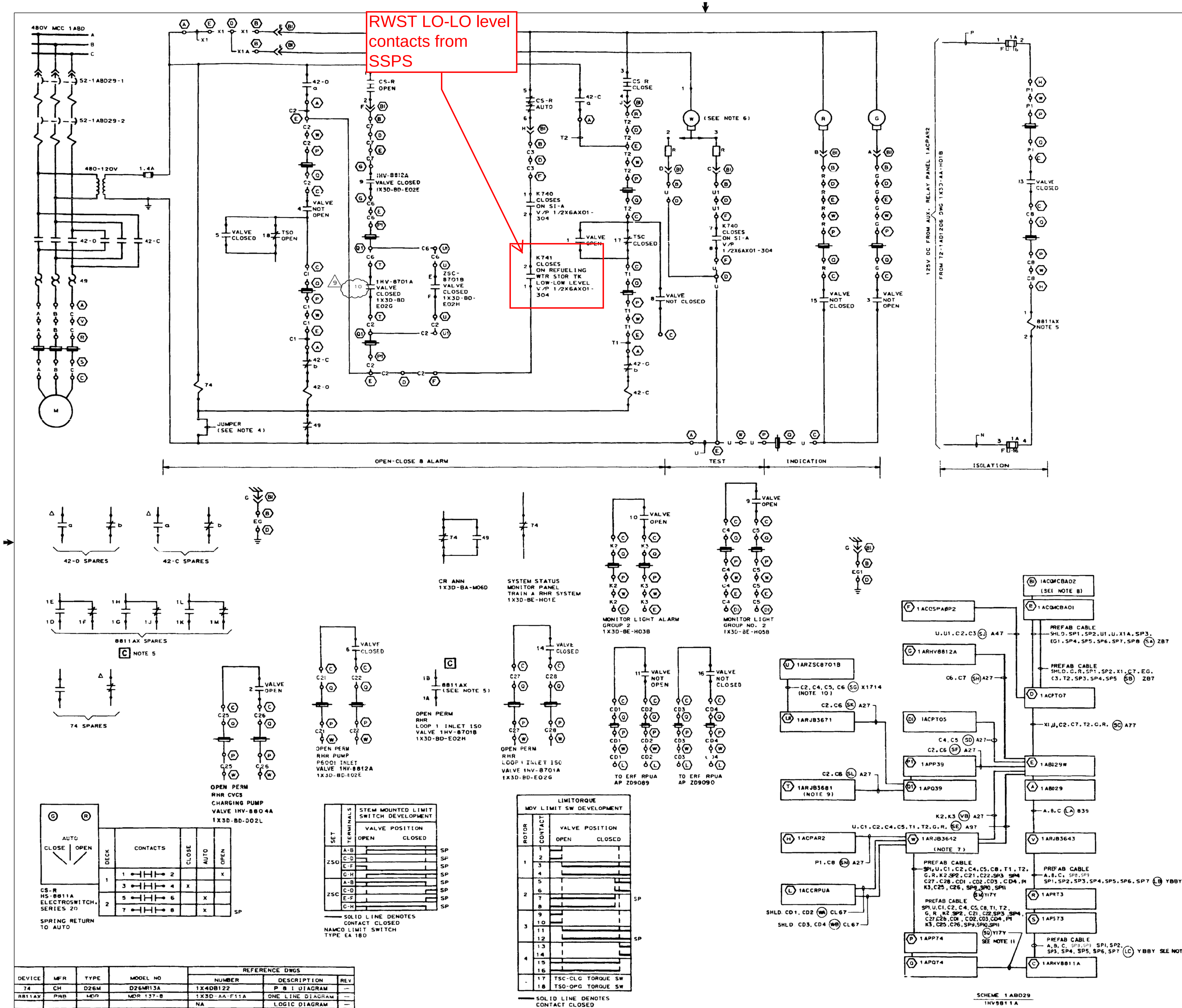
SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X4DB121 | VER. 42.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



480V MCC 1ABD
52-1ABD41
480-120V
42-O
42-C
49
74
JUMPER (SEE NOTE 4)
CS-R OPEN
CS-R CLOSE
VALVE CLOSED
TSO OPEN
VALVE NOT OPEN
VALVE OPEN
TSC CLOSED
VLV NOT CLOSED
VALVE NOT CLOSED
OPEN-CLOSE & ALARM
INDICATION
42-O SPARES
42-C SPARES
74 SPARES
CR ANN 1X3D-BA-M06D
MONITOR LIGHT ALARM GROUP NO.2-TRAIN A 1X3D-BE-H03B
MONITOR LIGHT GROUP NO. 2 1X3D-BE-H05B
SYS STATUS MONITOR PNL TRAIN A CTMT SPRAY SYS 1X3D-BE-H01G
TO ERF RPUA AP Z09410
TO ERF RPUA AP Z09411
Open circuitry does not contain any SSPS contacts. Manual open only.
Sump Suction HS does not have an AUTO position.
CLOSE OPEN
CONTACTS
CS-R HS-9017A ELECTROSWITCH SERIES 20
SPRING RETURN TO NORMAL
LIMITORQUE MOV LIMIT SW DEVELOPMENT
VALVE POSITION OPEN CLOSED
TSC-CLG TORQUE SW
TSO-OPG TORQUE SW
SOLID LINE DENOTES CONTACT CLOSED
REFERENCE DWGS
SCHEME 1ABD41 1HV9017A S.U. NO. BN
NOTES:
1. FOR GENERAL NOTES AND SYMBOLS REFER TO DWGS. LISTED ON 1X3D-AA-A00B.
2. ◯ DENOTES EQUIPMENT LOCATION NUMBER FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH "△" IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP & PERIODIC TESTING.
NUCLEAR SAFETY RELATED A
CONTAINMENT SPRAY PUMP P6001 RWST INLET
SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
ELEMENTARY DIAGRAM
CONTAINMENT SPRAY SYSTEM
1HV-9017A
SCALE: NONE
DRAWING NO.
1X3D-BD-J02G
REV.
6
JOB NO. 10604



NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DWGS. LISTED ON 1X3D-AA-A008.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH A "Δ" IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP & PERIODIC TESTING.
5. RELAY 8811AX IS AN ISOLATION RELAY. COIL CKT IS TRAIN A & OUTPUT CONTACT CKT'S ARE TRAIN C.
6. THE TEST LIGHT SHOULD BE LOCATED NEAR THE SAFETY INJECTION RWST RESET SWITCH WHICH IS SHOWN ON THE SOLID STATE INTERCONNECTION DIAGRAM. PART OF 2LB-40135
7. JUNCTION BOX 1ARJB3642 IS ALSO SHOWN IN DWG 1X3D-BD-D02L. E02E & G.
8. HS-8811A IS LOCATED AT B1. INTERNAL CONNECTION BETWEEN B1 AND B IS BY WESTINGHOUSE. THE PREFAB CABLE CONNECTOR IS LOCATED AT B.
9. ALSO SHOWN ON DWG. 1X3D-BD-D02L & E02G.
10. WIRE NOS C4 & C5 ARE SHOWN ON 1X3D-BD-D02L.
11. CONDUCTORS NOS. 1,13,16,17 SHOWN SPARE ARE DAMAGED NOT TO BE USED IN FUTURE.
12. CONDUCTORS NOS. 6,7,8 SHOWN SPARE ARE DAMAGED NOT TO BE USED IN FUTURE.

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	DESCRIPTION	REV
74	CH	D26M	D26MR13A	1X4DB122	P & I DIAGRAM	—
8811AX	P&B	MDR	MDR 137-8	1X3D-AA-F11A	ONE LINE DIAGRAM	—
				NA	LOGIC DIAGRAM	—
				1/2X6AF01-33	BECHTEL V/P	—

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
9	INCORP. PER ABN-33967	03-22-99	ELC	EOG	JGM	
0	ISSUED FOR CONSTRUCTION	SEE MICROFILM FOR SIGNATURES				

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
RESIDUAL HEAT REMOVAL SYSTEM
1HV-8811A

SCALE: NONE | JOB NO. 10604 | DRAWING NO. 1X3D-BD-E03F | REV. 9

SIZE E 34X44 | CAD NAME 13BDE03F

1. 026K2.02 001/LOIT/RO/C/A 2.7/2.9/026K2.02/LO-PP-11101-55///062K1.02

Given the following:

- An SI occurred and has **NOT** been reset.
- An LOSP occurs a few minutes later.
- 1AA02 is powered from DG1A.
- 1BA03 is powered from DG1B.

While the DGs are operating, an electrical perturbation results in the following:

- DG1A 186A lockout relay energizes (Generator Differential).
- DG1B 186B lockout relay energizes (Phase Overcurrent).

Which one of the following describes the current condition of the Containment Spray discharge valves?

1HV-9001A, Containment Spray Pump 'A' Discharge Isolation, is __(1)__,

and

1HV-9001B, Containment Spray Pump 'B' Discharge Isolation, is __(2)__.

	__(1)__	__(2)__
A.	energized	energized
B✓	de-energized	energized
C.	energized	de-energized
D.	de-energized	de-energized

K/A

026 Containment Spray

K2.02 Knowledge of bus power supplies to the following:

- MOVs

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of power supplies which feed the Containment Spray MOVs by requiring the candidate to determine if the MOV will be energized following 186 protective relay actuations concurrent with SI.

EXPLANATION OF REQUIRED KNOWLEDGE

Per SOP 13145A/B-1/2 Precautions and Limitations, with SI present and a 186A relay actuation, the associated DG output breaker will open and the DG will shutdown. With SI present and a 186B relay actuation, the associated DG output breaker will remain closed and the DG will continue to operate. Therefore, 1AA02 will de-energize and 1BA03 will remain energized.

The one-line power for 1HV9001A is 4160V SWGR 1AA02 -> 480V SWGR 1AB15 -> 480V MCC 1ABD.

The one-line power for 1HV9001B is 4160V SWGR 1BA03 -> 480V SWGR 1BB16 -> 480V MCC 1BBD.

[Note - The valve titles used are from 13115-1 - not the QMCB (they are not worded the same).]

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The Generator Differential (186A) lockout relay will emergency trip DG1A under all conditions, de-energizing bus 1AA02, and removing power from 1HV-9001A.. However, 186 B and C lockout relays also exist, each causing the DG and output breaker to behave differently based on the presence of an LOSP and/or a SI signal. These lockouts are easily confused. This distractor would be correct for either 186 B or C lockout relay.

The second part is correct. With SI present and a 186B relay actuation, the 1B DG output breaker will remain closed and the DG will continue to operate.

B. Correct. The first part is correct. With SI present and a 186A relay actuation, the 1A DG output breaker will open and the DG will shut down.

The second part is correct. See the second part of choice A above.

C. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Phase Overcurrent (186B) would not trip DG1B or its output breaker because this trip is not active during SI conditions. However, 186 A and C lockout relays also exist, each causing the DG and output breaker to behave differently based on the presence of an LOSP and/or a SI signal. These lockouts are easily confused. This distractor would be correct for a 186A lockout relay.

D. Incorrect. Plausible. The first part is correct. See the first part of choice B above.

The second part is incorrect. See the second part of choice C above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	026K2.02
Importance Rating:	2.7 / 2.9

Technical Reference: 13145A-1 Rev 6, page 13
1X3D-AA-F12A Rev 18.0
1X3D-AA-F12B Rev 6.0
1X3D-AA-E17A Rev 12.0
1X3D-AA-F11A Rev 24.0
1X3D-AA-F11B Rev 10.0
1X3D-AA-E16A Rev 9.0

References provided: None

Learning Objective: LO-PP-11101-55 Identify the primary relays which will actuate each of the following lockout relays and how the diesel generator will respond to each normal start from SI, UV, Local Emergency Start, and Normal Start.

a. 186A lockout relay
b. 186B lockout relay

LO-TA-01023 Draw the electrical distribution system

Question origin: MODIFIED - HL18 NRC Question # 062K1.02

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7

Comments: Question appears to match the KA. Answer choices could be cleaned up by asking:

1HV-9001A is __(1)___ and 1HV-9001B is __(2)___.

Then all choices can toggle between energized/de-energized. Question appears to be okay.
-JAT 12/19/2013 (Editorial)

Comment incorporated. SAT
-JAT 2/4/14

You have completed the test!

1. 062K1.02 001/2/1/AC DIST - EDG/C/A - 2.5/MODIFIED/R/NRC RO/TNT / RLM

Given the following:

- An SI has occurred and is **NOT** reset.
- An LOSP occurs a few minutes later.
- 1AA02 is powered from DG1A
- 1BA03 is powered from DG1B

While the DG's are operating, an electrical perturbation results in the following;

- DG1A 186A lockout relay energizes (Generator Differential)
- DG1B 186B lockout relay energizes (Phase Overcurrent)

Which **ONE** of the following is **CORRECT** with respect to the status of power to the 4160 1E Emergency Buses at this time?

A. Both 4160 1E emergency buses would be energized.

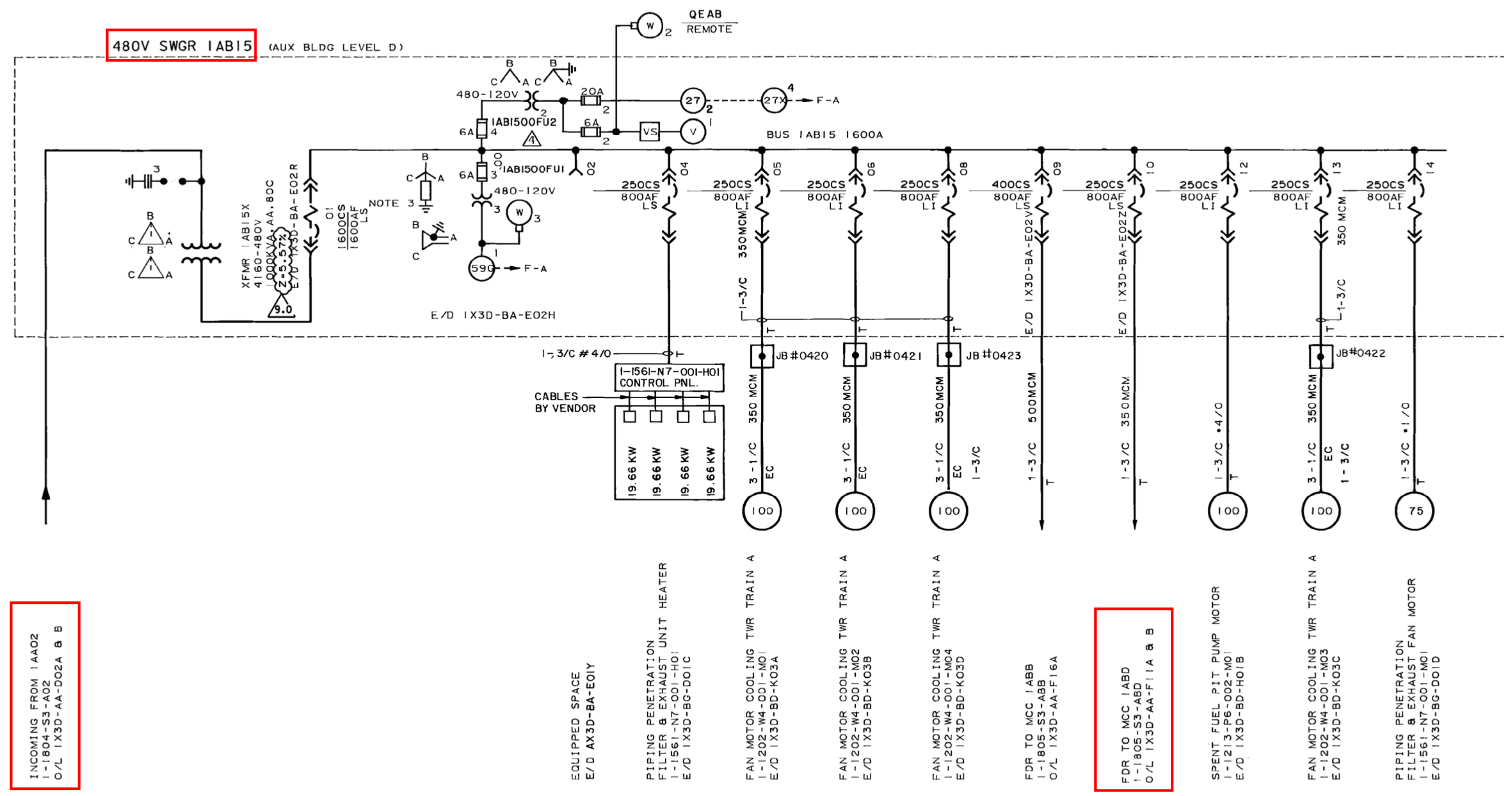
B. Both 4160 1E emergency buses would be de-energized.

C. 1AA02 would be energized, 1BA03 would be de-energized.

D.✓ 1BA03 would be energized, 1AA02 would be de-energized.

2.2.18 The following table lists the DG Lockout Relays and the related functions:

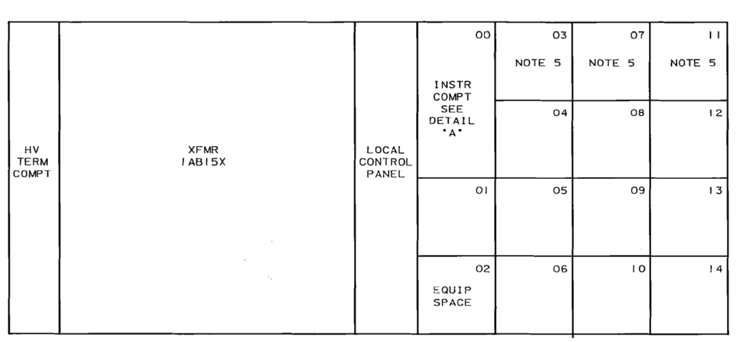
LOCKOUT RELAYS	PRIMARY RELAYS	BREAKER STATUS	ENGINE STATUS
186A	Generator Differential	**Trips Open** Always	**Shuts Down** Always
186B	Phase Overcurrent or Loss of Field	**Trips Open** on Normal Start, Local Emergency Start, or LOSP	**Shuts Down** on Normal Start
		Remains Closed on SI	**Remains Running** on LOSP, Local Emergency Start, or SI
186C	Reverse Power or Negative Phase Sequence	**Trips Open** in Parallel Mode Only	**Remains Running**



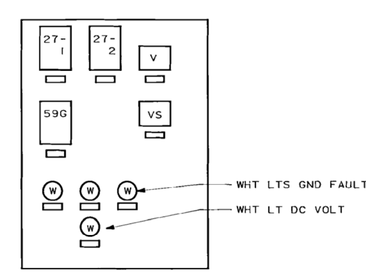
LEGEND				
DEVICE NO	MFR	TYPE	DESCRIPTION	MOD OR STYLE NO
27-1 27-2	GE	IAV	UNDERVOLTAGE RELAY, 115V AC 55-140V OPERATING RANGE	121AV53K1A
27X1-1,2 27X2-1,2	GE	HFA	AUX TRIP RELAY, 125V DC	12HFA51A42H
59	GE	IAV	OVERVOLTAGE RELAY, 199V AC 16-64V OPERATING RANGE	121AV51D1A
LI	ABB	PR112	MICROPROCESSOR TRIP DEVICE WITH LONGTIME & INSTANTANEOUS TRIP	PR112/P-A
LS	ABB	PR112	MICROPROCESSOR TRIP DEVICE WITH LONGTIME & SHORT TIME TRIPS	PR112/P-A
W			INDICATING LIGHT RESISTOR TYPE WHITE LENS	
VS			VOLTMETER SWITCH	
V			VOLTMETER	

NOTES:

1. FOR DEVICE TABLE, GENERAL NOTES, REFERENCE DWG'S & LEGEND SEE 1X3D-AA-A00A THRU 1X3D-AA-A00E.
2. 250CS ← CURRENT SENSOR RATING
   800AF ← SWITCHGEAR FRAME SIZE
3. SIZE OF STABILIZING RESISTOR IS 25 OHM.
4. THE TYPE AND MODEL NUMBER OF DEVICES SHOWN ON THIS DWG ARE FOR FUNCTIONAL DESCRIPTION ONLY. THE VENDOR IS RESPONSIBLE FOR THEIR QUALIFICATION AS CLASS 1E DEVICES.
5. SPACE RESERVED FOR TERMINAL BLOCKS AND AUXILIARY RELAYS.
6. ANY CHANGES, WHICH AFFECT THE LARGEST LOAD (ADDING OR REMOVING THE LOAD OR CHANGING THE BREAKER SIZE), MUST BE REVIEWED FOR ACCEPTABILITY IN CALCULATION X3CT08



480V SWGR 1AB15
FRONT ELEVATION
(NOT TO SCALE)



DETAIL "A"

FUNCTIONAL TABLE	
FUNCTION CODE-F	DESCRIPTION
A	ANNUNCIATOR

NUCLEAR SAFETY RELATED A

NO.	VERSIONS	DATE	DR	CHK	APPV
9.0	REVISED PER ABN-V02294, VER. 1.0	08/27/10	TSL	WLM	JMR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

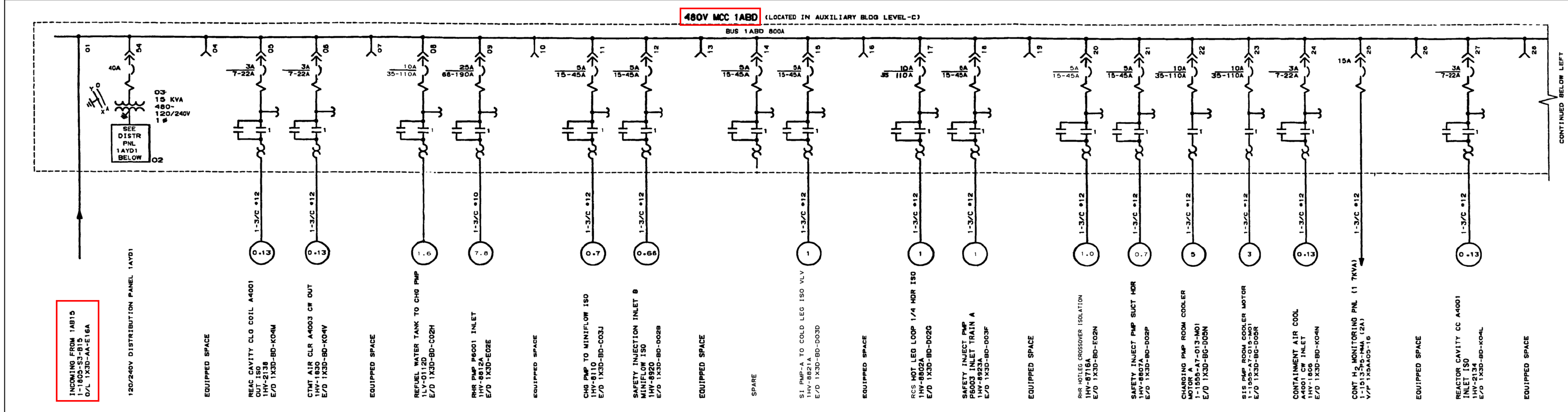
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

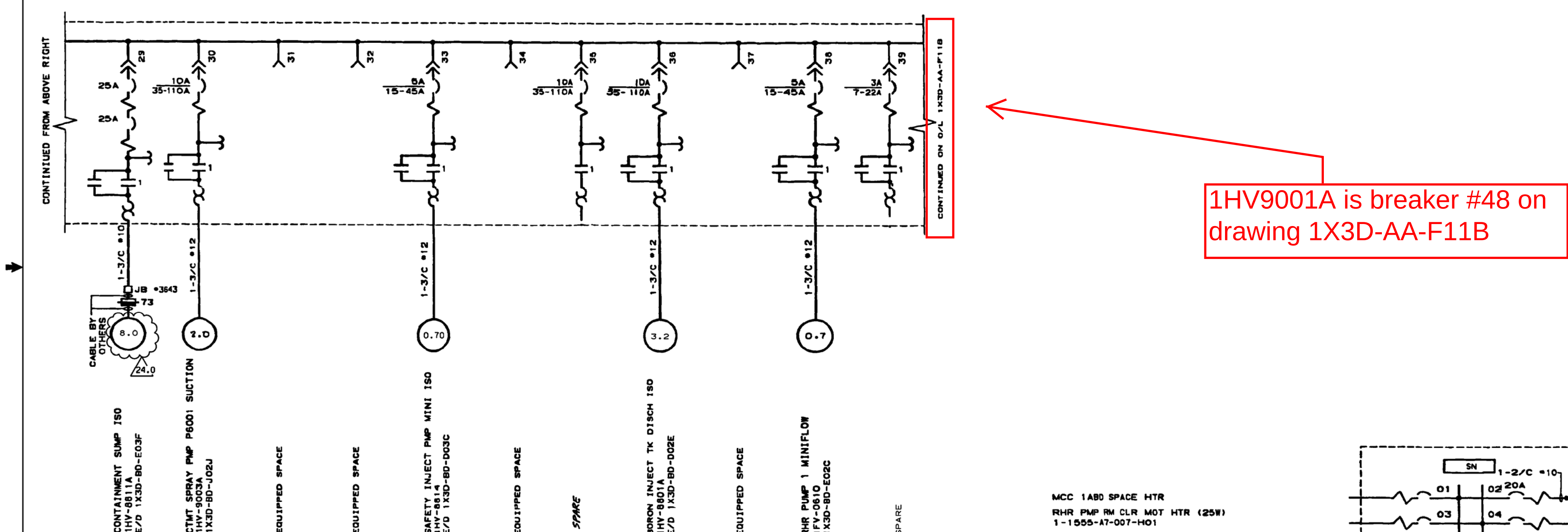
ONE LINE DIAGRAM
480V SWITCHGEAR 1AB15
1-1805-S3-B15

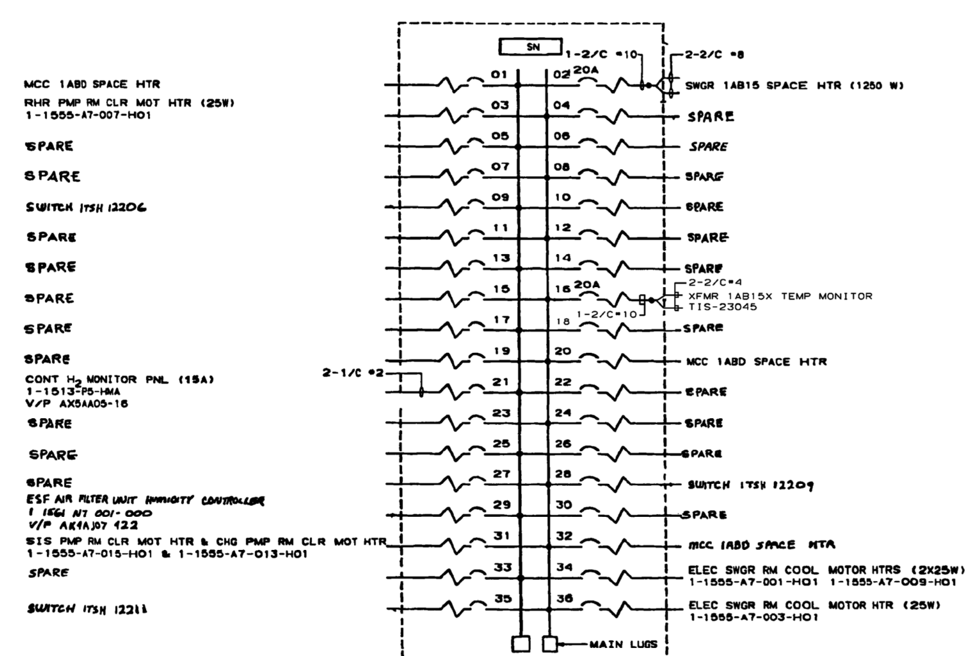
SCALE: 1/8"=1'-0"	DRAWING NO.	VER.
JOB NO. 10604	1X3D-AA-E16A	9.0

SIZE E 34X44
DRAWING CATEGORY: CRITICAL

480V MCC 1ABD (LOCATED IN AUXILIARY BLDG LEVEL-C)







PANEL 1AYD1
120/240V 1φ 3 WIRE
(FOR CABLE SIZES SEE NOTE 6)
(FOR PANEL LOADING SEE CALC. X3CH03)

NOTES:

1. FOR DEVICE TABLE GENERAL NOTES REFERENCE DWGS. AND LEGEND SEE O/L 1X3D-AA-A00A THRU 1X3D-AA-A00E.
2. UNLESS OTHERWISE NOTED ALL 480V BREAKERS ARE 150A FRAME SIZE.
3. ALL 120/240V PANEL CIRCUIT BREAKERS ARE 1P WITH 15 AMP TRIP RATING.
4. BREAKER RATING (TYPICAL)
   10A → CONTINIOUS RATING (SHOWN FOR ALL LOADS)
   35-110A → ADJUSTABLE LOW AND HIGH MAGNETIC TRIP RANGE (SHOWN FOR ALL MOTOR BREAKERS)
5. FOR FRONT ELEVATION SEE O/L 1X3D-AA-F11B.
6. ALL PANEL CABLES ARE 2/C #12 UNLESS OTHERWISE NOTED.
7. BACK-UP BREAKER SHALL BE THERMAL MAGNETIC.
8. ANY CHANGES, WHICH AFFECT THE LARGEST LOAD (ADDING OR REMOVING THE LOAD OR CHANGING THE BREAKER SIZE) MUST BE REVIEWED FOR ACCEPTABILITY IN CALCULATION X3CT08.

NO.	VERSIONS	DATE	DR	CHK	APPV
24.0	REVISED PER ABN 1081089101E002, VER 1.0	10-02-12	DCP	ELC	JMR

NUCLEAR SAFETY RELATED A Q

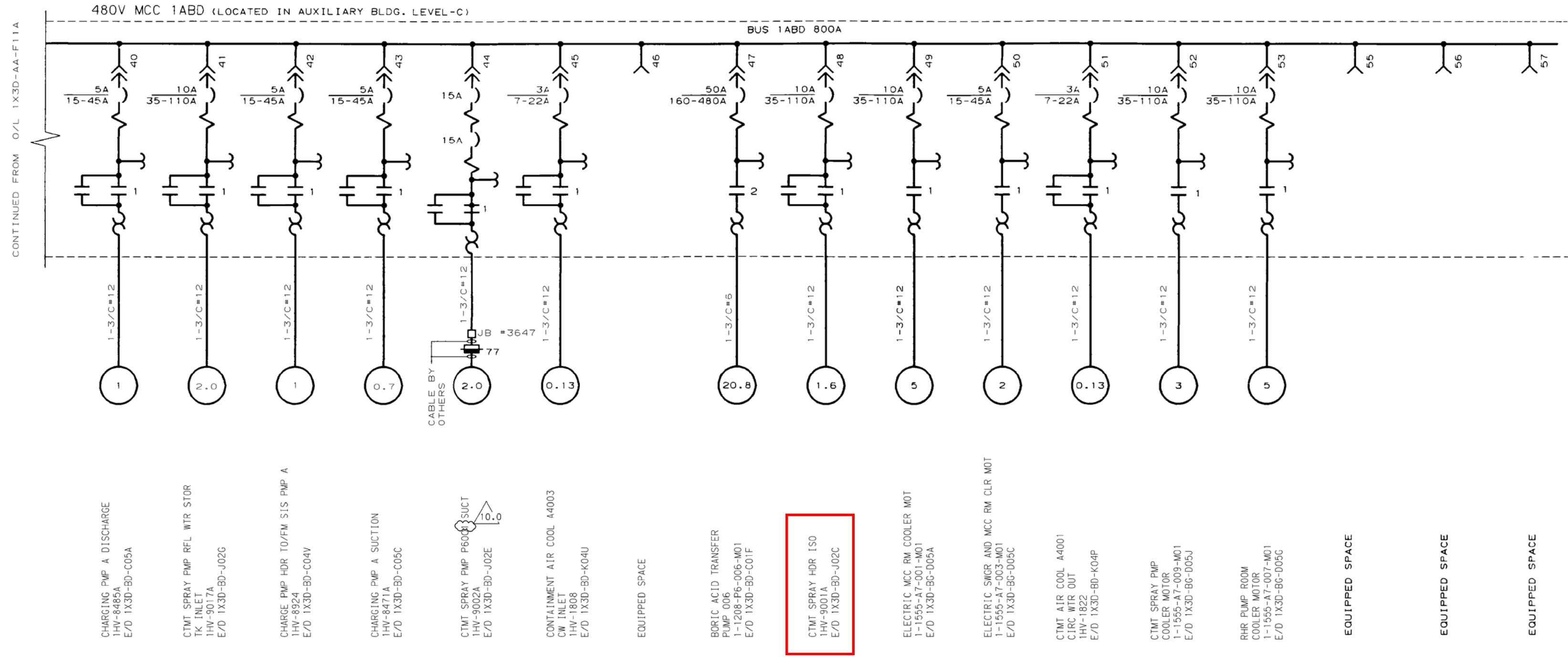
SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

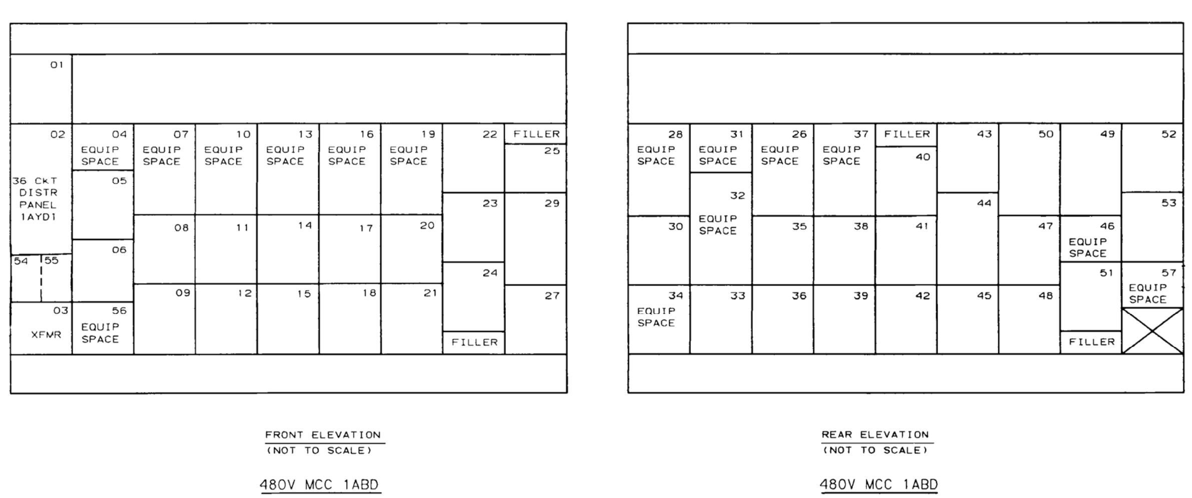
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ONE LINE DIAGRAM
480V MOTOR CONTROL CENTER 1ABD
1-1805-S3-ABD

SCALE: NONE	DRAWING NO.	VER.
JOB NO. 10604	1X3D-AA-F11A	24.0

SIZE E 34X44
DRAWING CATEGORY: CRITICAL





NOTES:
1. REFER TO DWG. 1X3D-AA-F11A.

NUCLEAR SAFETY RELATED A

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

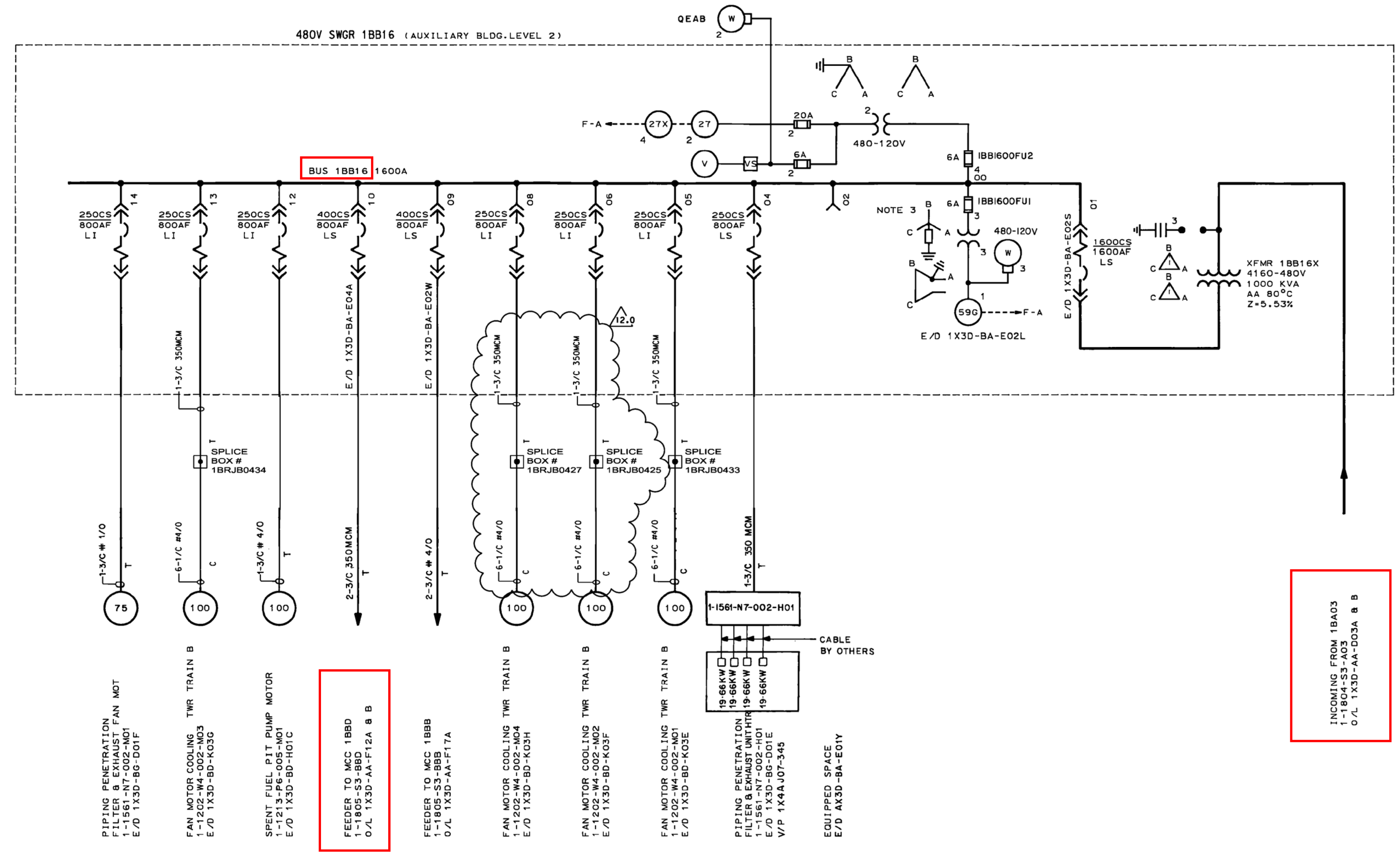
ONE LINE DIAGRAM
480V MOTOR CONTROL CENTER 1ABD
1-1805-S3-ABD

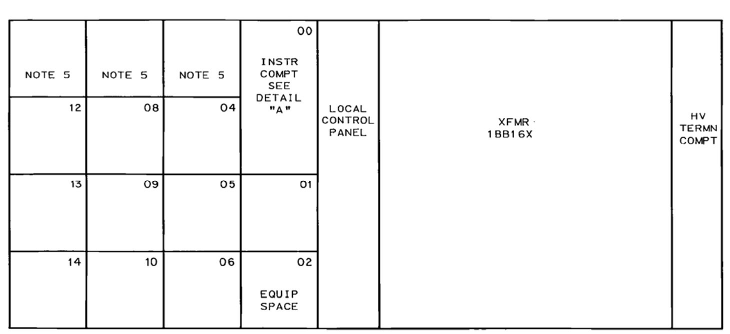
SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-F11B	10.0

NO.	VERSIONS	DATE	DR	CHK	APPV
10.0	REVISED PER ABN-V00837, VER. 1.0	07-16-07	DFV	JEM	WLN

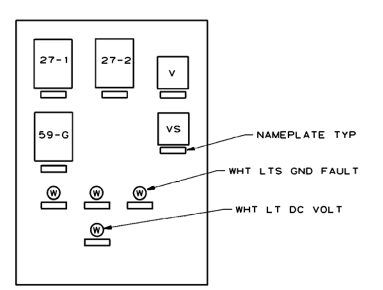
SIZE E 34X44

DRAWING CATEGORY: CRITICAL





480V SWGR 1BB16
FRONT ELEVATION
(NOT TO SCALE)



DETAIL "A"

LEGEND				
FUNCTION NO.	MFR	TYPE	DESCRIPTION	MOD OR STYLE NO.
27-1 27-2	GE	IAV	UNDERVOLTAGE RELAY, 115V AC 55-140V OPERATING RANGE	12IAV53K1A
27X1-1,2 27X2-1,2	GE	HFA	AUX TRIP RELAY, 125V DC	12HFA51A42H
59G	GE	IAV	OVERVOLTAGE RELAY 199V AC 16-64V OPERATING RANGE	12IAV51D1A
LI	ABB	PR112	SOLID STATE TRIP DEVICE WITH LONGTIME & INSTANTANEOUS TRIP	PR112/P-A
LS	ABB	PR112	SOLID STATE TRIP DEVICE WITH LONGTIME & SHORT TIME TRIPS	PR112/P-A
(W)			INDICATING LIGHT RESISTOR TYPE WHITE LENS	

FUNCTIONAL TABLE	
FUNCTION CODE-F	DESCRIPTION OF DEVICE FUNCTION
A	ANNUNCIATOR
B	TRIP SELECTED BREAKERS

NOTES:

1. FOR DEVICE TABLE, GENERAL NOTES, REFERENCE DWGS & LEGEND SEE 1X3D-AA-A00A THRU 1X3D-AA-A00E.
2. 250CS ← CURRENT SENSOR RATING; 800AF ← SWITCHGEAR FRAME RATING
3. 25 OHM STABILIZING RESISTOR.
4. THE TYPE AND MODEL NUMBER OF DEVICES SHOWN ON THIS DWG ARE FOR FUNCTIONAL DESCRIPTION ONLY. THE VENDOR IS RESPONSIBLE FOR THEIR QUALIFICATION AS CLASS 1E DEVICES.
5. SPACE RESERVED FOR TERMINAL BLOCKS AND AUXILIARY RELAYS.
6. ANY CHANGES, WHICH AFFECT THE LARGEST LOAD (ADDING OR REMOVING THE LOAD OR CHANGING THE BREAKER SIZE), MUST BE REVIEWED FOR ACCEPTABILITY IN CALCULATION X3CT08

NUCLEAR SAFETY RELATED B Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

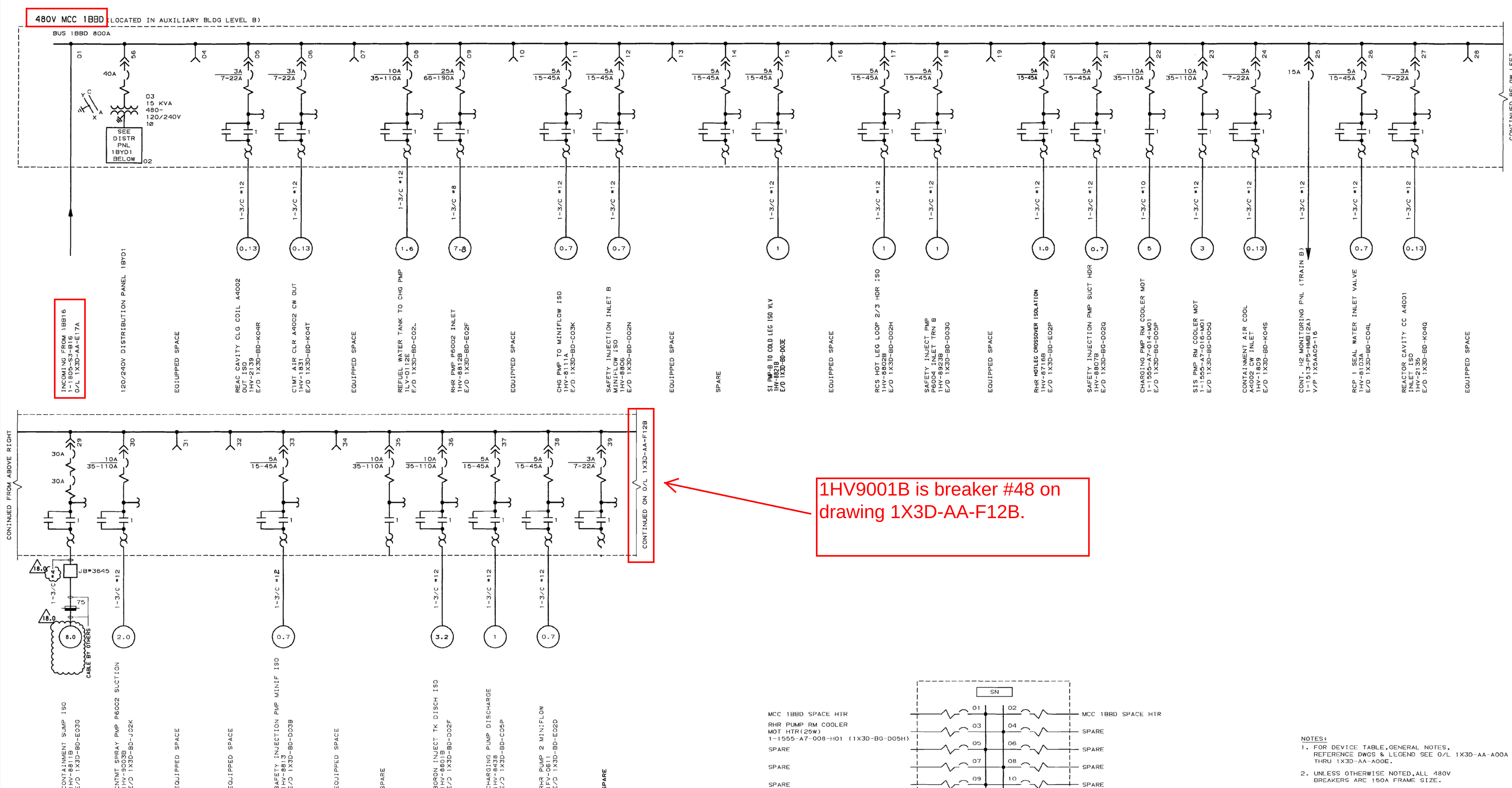
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

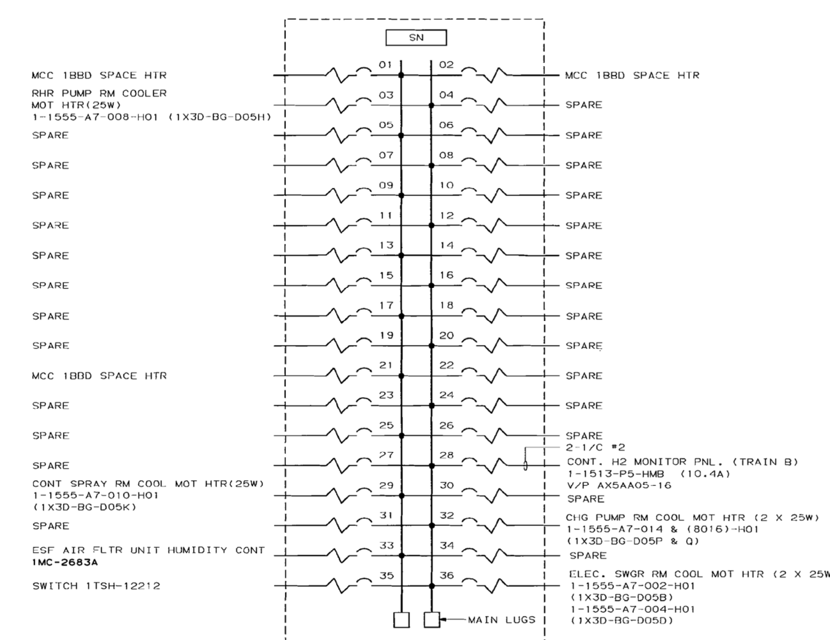
ONE LINE DIAGRAM
480V SWITCHGEAR 1BB16
1-1805-S3-B16

NO.	VERSIONS	DATE	DR	CHK	APPV
12.0	REVISED PER ABN 1081982101E003, VER. 2.0	11/21/11	JLO	GDM	JMR

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-E17A	12.0

SIZE E 34X44
DWG CATEGORY: CRITICAL





PANEL 1BYD1

120/240V, 1Ø, 3 WIRE
(FOR CABLE SIZES SEE NOTE 5)

(FOR PANEL LOADING SEE CALC. X3CH03)

NOTES:

1. FOR DEVICE TABLE, GENERAL NOTES, REFERENCE DWGS & LEGEND SEE O/L 1X3D-AA-A00A THRU 1X3D-AA-A00E.
2. UNLESS OTHERWISE NOTED, ALL 480V BREAKERS ARE 150A FRAME SIZE.
3. ALL 120/240V PANEL CIRCUIT BREAKERS ARE 1P WITH 15AMP TRIP RATING.
4. BREAKER RATING (TYPICAL)
   10A → CONTINUOUS RATING (SHOWN FOR ALL LOADS)
   35-110A → ADJUSTABLE LOW AND HIGH MAGNETIC TRIP RANGE (SHOWN FOR ALL MOTOR BREAKERS)
5. UNLESS OTHERWISE SPECIFIED, ALL PANEL CABLES ARE 2/C #12.
6. ANY CHANGES, WHICH AFFECT THE LARGEST LOAD (ADDING OR REMOVING THE LOAD OR CHANGING THE BREAKER SIZE), MUST BE REVIEWED FOR ACCEPTABILITY IN CALCULATION X3CT08.

NUCLEAR SAFETY RELATED B Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

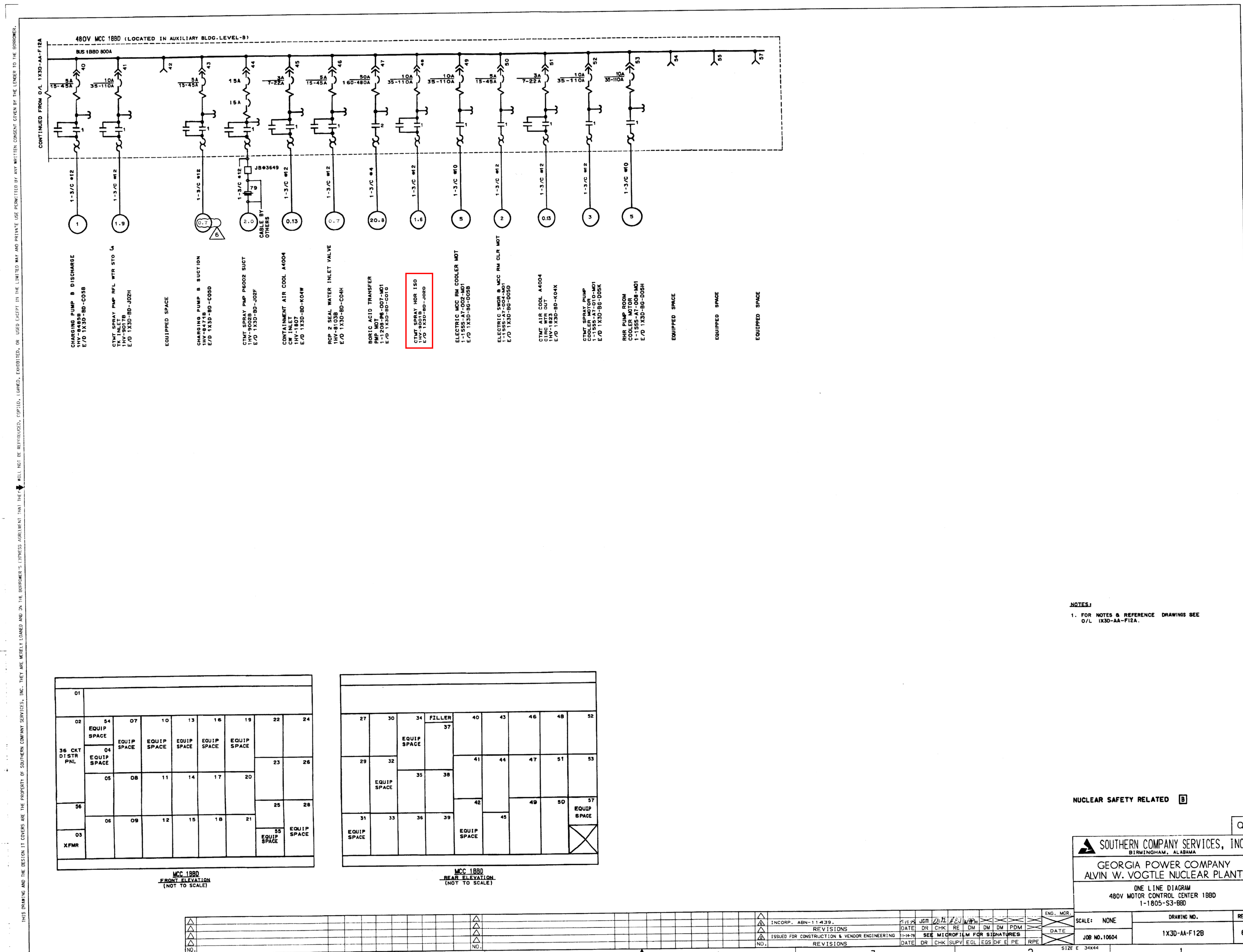
ONE LINE DIAGRAM
480V MOTOR CONTROL CENTER 1BBD
1-1805-S3-BBD

NO.	VERSIONS	DATE	DR	CHK	APPV
18.0	REV. PER ABN 1081089101E003, VER.2.0	3/22/11	JLO	WLM	JMR

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-F12A	18.0

SIZE E 34X44

DWG CATEGORY: CRITICAL


480V MCC 1BBD (LOCATED IN AUXILIARY BLDG.-LEVEL-B)
BUS 1BBD 800A
CONTINUED FROM O/L 1X3D-AA-F12A
CHARGING PUMP B DISCHARGE 1HV-8485B E/O 1X3D-BD-C05B
CTMT SPRAY PMP RFL WTR STO TK INLET 1HV-9017B E/O 1X3D-BD-J02H
EQUIPPED SPACE
CHARGING PUMP B SUCTION 1HV-8471B E/O 1X3D-BD-C05D
CTMT SPRAY PMP P6002 SUCT 1HV-9003B E/O 1X3D-BD-J02F
CABLE BY OTHERS
JB#3649
CONTAINMENT AIR COOL A4004 CW INLET 1HV-1807 E/O 1X3D-BD-K04W
RCP 2 SEAL WATER INLET VALVE 1HV-8103B E/O 1X3D-BD-C04H
BORIC ACID TRANSFER PMP MOT 1-1208-P6-007-M01 E/O 1X3D-BD-C01G
CTMT SPRAY HDR ISO 1HV-9001B E/O 1X3D-BD-J02D
ELECTRIC MCC RM COOLER MOT 1-1555-A7-002-M01 E/O 1X3D-BG-D05B
ELECTRIC SWGR & MCC RM CLR MOT 1-1555-A7-004-M01 E/O 1X3D-BG-D05D
CTMT AIR COOL A4004 CIRC WTR OUT 1HV-1823 E/O 1X3D-BD-K04X
CTMT SPRAY PUMP COOLER MOTOR 1-1555-A7-010-M01 E/O 1X3D-BG-D05K
RHR PUMP ROOM COOLER MOTOR 1-1555-A7-008-M01 E/O 1X3D-BG-D05H
EQUIPPED SPACE
EQUIPPED SPACE
EQUIPPED SPACE
NOTES:
1. FOR NOTES & REFERENCE DRAWINGS SEE O/L 1X3D-AA-F12A.
MCC 1BBD FRONT ELEVATION (NOT TO SCALE)
MCC 1BBD REAR ELEVATION (NOT TO SCALE)
NUCLEAR SAFETY RELATED B
SOUTHERN COMPANY SERVICES, INC. BIRMINGHAM, ALABAMA
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
ONE LINE DIAGRAM
480V MOTOR CONTROL CENTER 1BBD
1-1805-S3-BBD
SCALE: NONE
JOB NO. 10604
DRAWING NO. 1X3D-AA-F12B
REV. 6
INCORP. ABN-11439.
ISSUED FOR CONSTRUCTION & VENDOR ENGINEERING
REVISIONS
SEE MICROFILM FOR SIGNATURES

1. 027AK2.03 001/LOIT/RO/C/A 2.6/2.8/027K2.03////

Initial conditions:

- Unit 1 is at 100% reactor power.
- One group of pressurizer backup heaters is energized in MANUAL.
- Pressurizer pressure control is selected to CH 455 / 456.

Current condition:

- Pressurizer pressure transmitter, 1PT-455, fails LOW.

Which one of the following completes the following statement?

In response to the failure, the remaining pressurizer backup heaters will __(1)__,

and

the pressurizer spray valve controllers will demand full __(2)__ position.

	__(1)__	__(2)__
A.	energize	open
B.✓	energize	closed
C.	remain off	open
D.	remain off	closed

K/A

027 **Pressurizer Pressure Control System Malfunction**

K2.03 **Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:**

- Controllers and positioners

K/A MATCH ANALYSIS

The question requires the candidate to determine the response of the pressurizer heaters and the spray controller demand to a loss of a pressurizer pressure transmitter.

EXPLANATION OF REQUIRED KNOWLEDGE

The pressurizer pressure control circuit utilizes primary and secondary selectable control loops. With 455 / 456 selected, PT-455 is the input to the primary control loop. PT-455 failing low will result in the pressurizer master controller demand lowering, resulting in all heaters on. The associated pressurizer spray slave controller demand

will also lower, demanding full close on the spray valves. Reference 18000-C, Figure 1, for a diagram of the heater control circuit.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. PT-455 is the primary controlling channel which inputs to the heaters and the spray valves to maintain a constant 2235 psig. When PT-455 fails low, the control circuit will sense a low pressure condition and energize the heaters in response.

The second part is incorrect. PT-455 is the primary controlling channel which controls the spray valves to maintain a constant 2235 psig. When PT-455 fails low the control circuit will sense a low pressure condition and energize the heaters in response. However, a candidate without specific knowledge of the spray control circuit may assume that heaters and sprays are controlled by the primary and secondary loops respectively. In this case, PT-456 would see increasing pressure and call for an increase in spray flow.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. PT-455 is the primary controlling channel, which inputs to the spray valve controllers to maintain a constant 2235 psig. When PT-455 fails low the master controller will send a signal to the slave controllers to close the spray valves.

C. Incorrect. Plausible. The first part is incorrect. PT-455 is the primary controlling channel which inputs to the heaters and the spray valves to maintain a constant 2235 psig. When PT-455 fails low, the control circuit will sense the low pressure condition and energize the heaters in response. However, the candidate may confuse which channel inputs to the primary circuit and believe PT-456 is controlling. In this case, no change in heater condition is expected.

The second part is incorrect. When PT-455 fails low the control circuit will sense the low pressure condition and energize the heaters in response. However, a candidate may believe that, because the backup heater is energized in manual, the resulting failure will result in an actual high pressure condition. If the candidate confused which channel inputs to the primary circuit and believes PT-456 is controlling, he could determine that this would cause the spray valves to open to mitigate the pressure increase.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B

above.

Level: RO
Tier # / Group # T1 / G1
K/A# 027AK2.03
Importance Rating: 2.6 / 2.8

Technical Reference: LOGIC 1X6AA02-235, Rev 9.0
LOGIC 1X6AA02-236, Rev 7.0
AOP 18000-C, Rev 5.0, page 5

References provided: None

Learning Objective: LO-LP-60301-10 Given that the channel selector switch is in the NORMAL position (455/456), describe how and why the plant will respond to the following pressurizer pressure instrument failures. Consider each separately and include effects on the Pressurizer Pressure Control System response, alarms, RPS, and ESF actuations.

a. 455 fails high
b. 455 fails low
c. 456 fails high

Question origin: MODIFIED, HL-15 NRC Question # 027AK2.03

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 41.10 / 45.3

Comments:

You have completed the test!

1. 027AK2.03 001/1/1/PZR PRESS CNTRL/2.6/2.8 C/A/LORQ BANK/RO/SRO/NRC/GCW

Given the following:

- The unit is at 100% power.

- All control systems are in their normal alignments.

- The Pressurizer Master Pressure Controller output demand fails LOW.

Assuming no action has been taken by the crew, which ONE of the following describes the effect on the Pressurizer heaters, and the resulting effect on the plant?

A.✔ PZR heaters energize.

PZR pressure rise is controlled by PZR PORV operation.

B. PZR heaters energize.

PZR pressure rise is controlled by PZR spray valve operation.

C. PZR heaters de-energize.

ONLY the PZR spray valves open, reactor trips on low PZR pressure.

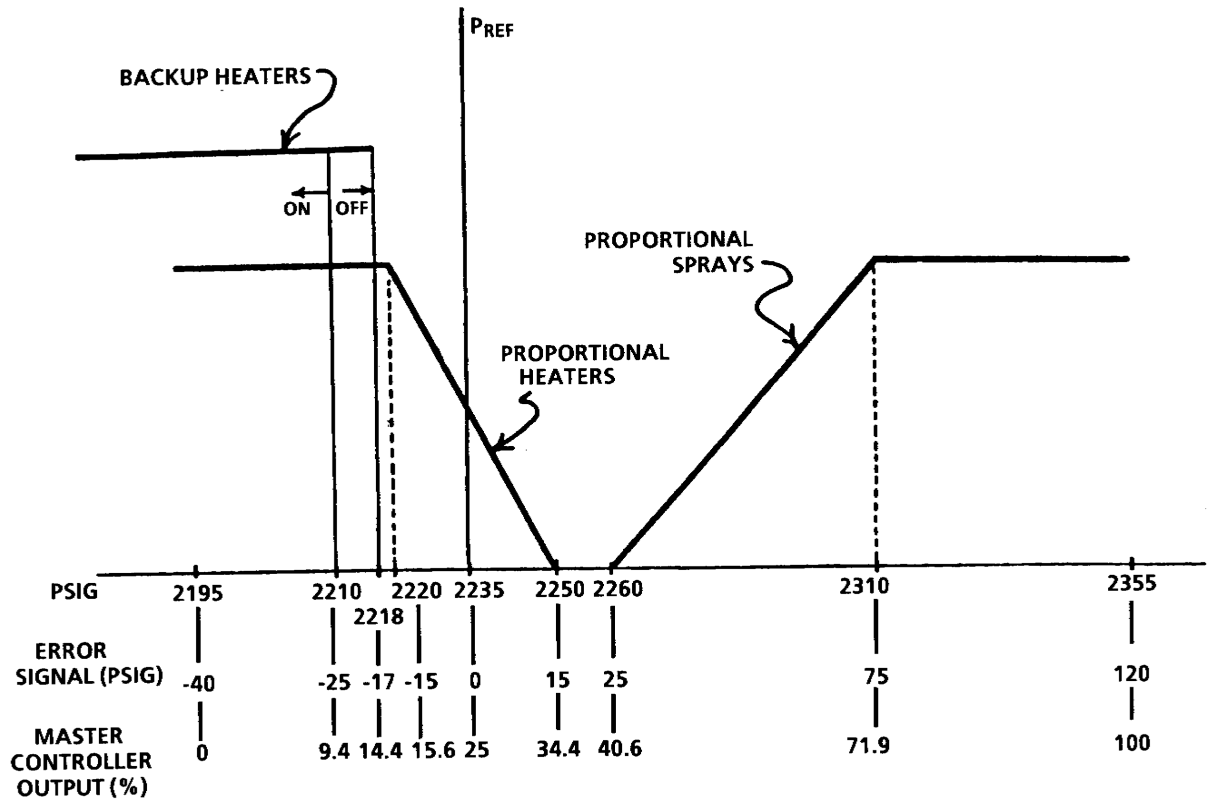
D. PZR heaters de-energize.

PZR spray valves and one PZR PORV open, reactor trips on low PZR pressure.

FIGURE 1

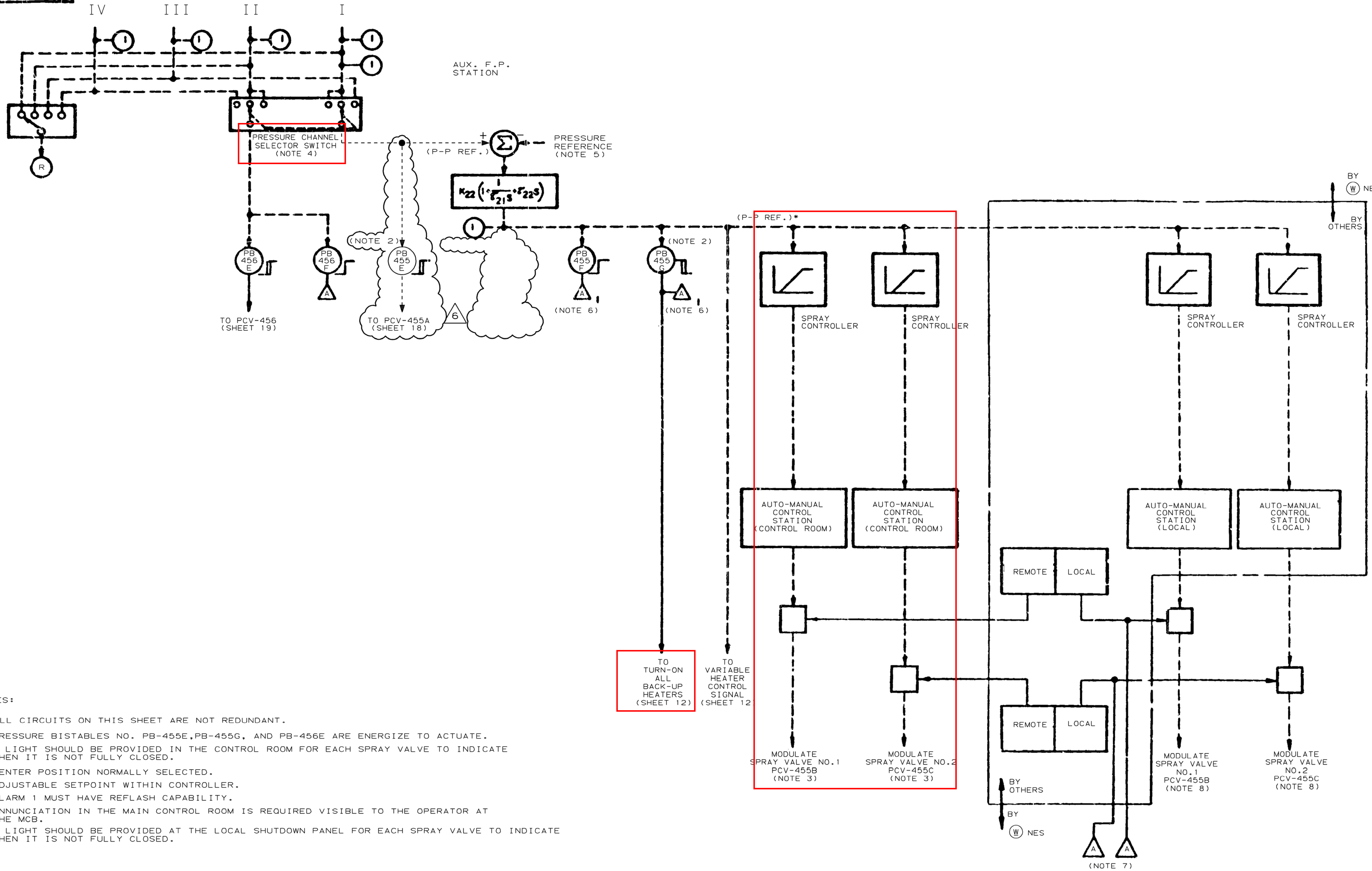
Sheet 1 of 1

PRESSURIZER PRESSURE CONTROLLER BAND



7243D07 SHEET-11

PRESSURIZER PRESSURE CHANNELS



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. PRESSURE BISTABLES NO. PB-455E,PB-455G, AND PB-456E ARE ENERGIZE TO ACTUATE.
3. A LIGHT SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN IT IS NOT FULLY CLOSED.
4. CENTER POSITION NORMALLY SELECTED.
5. ADJUSTABLE SETPOINT WITHIN CONTROLLER.
6. ALARM 1 MUST HAVE REFLASH CAPABILITY.
7. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED VISIBLE TO THE OPERATOR AT THE MCB.
8. A LIGHT SHOULD BE PROVIDED AT THE LOCAL SHUTDOWN PANEL FOR EACH SPRAY VALVE TO INDICATE WHEN IT IS NOT FULLY CLOSED.

NO.	REVISIONS	DATE	DR	CHK	APPV
6	INC. PER DCP 98-V1N0061	11-06-00	CD	EOG	MWD

SCS REVISIONS

DOCUMENT STATUS CODE 1

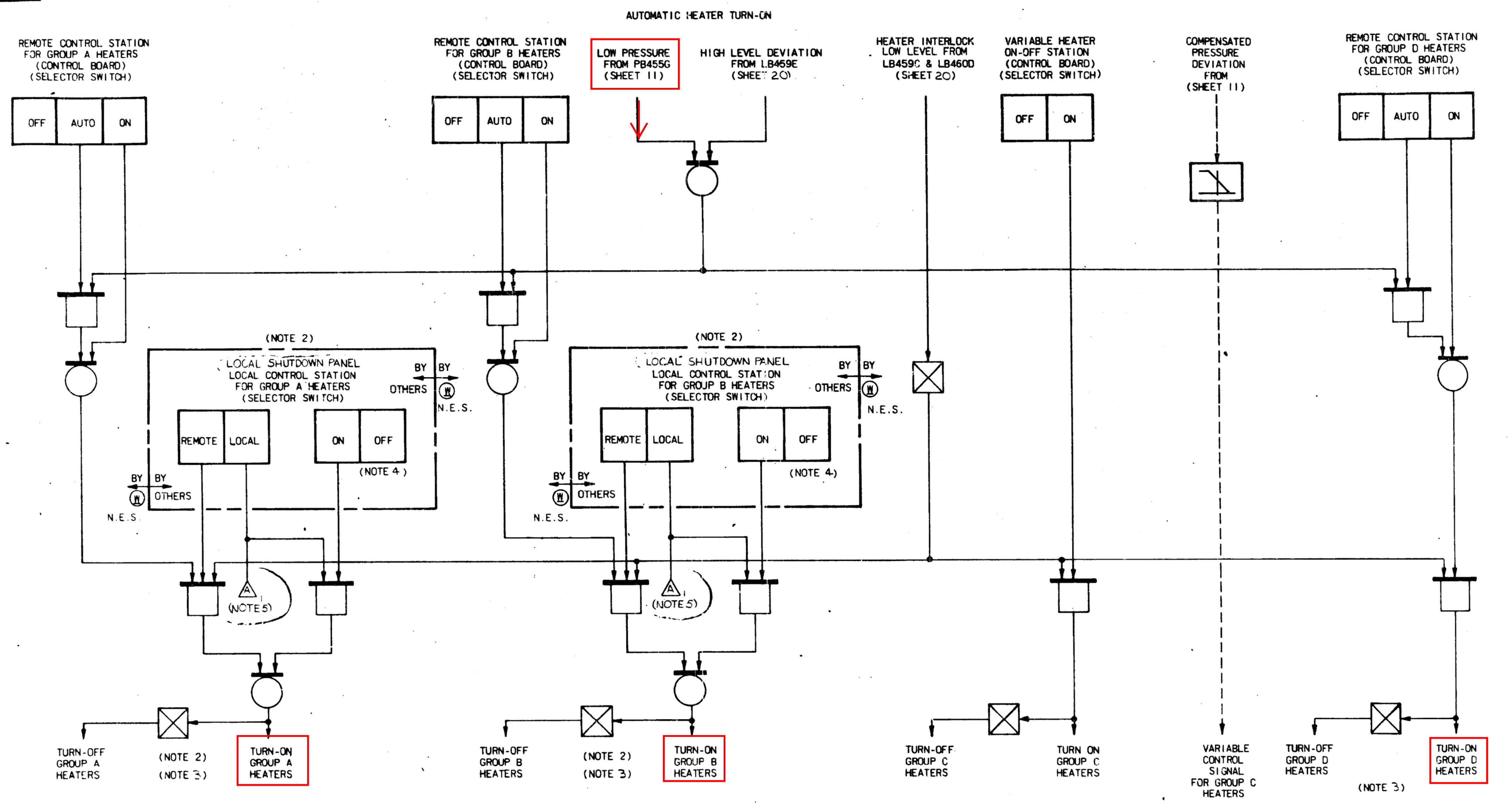
1X6AA02-00235-9

Westinghouse Electric Corporation

GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
PRESSURIZER PRESSURE CONTROL

7243D07
SHEET-11

7243D07 SHEET - 12



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. GROUP A AND GROUP B HEATERS MUST BE ON SEPARATE VITAL POWER SUPPLIES WITH THE LOCAL CONTROL SEPARATED SO THAT ANY SINGLE FAILURE DOES NOT DEFEAT BOTH.
3. BACK UP HEATER STATUS INDICATION IN CONTROL ROOM.
4. PRECAUTIONS SHOULD BE TAKEN TO AVOID MANUAL HEATER OPERATION, WHICH WOULD CAUSE HEATER DAMAGE, IF THE WATER LEVEL UNCOVERS THE HEATERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED TO BE VISIBLE TO THE OPERATOR AT THE MAIN CONTROL BOARD.

GEORGIA POWER COMPANY
PLANT: ALVIN W. VOGTLE NUCLEAR PLANT
UNITS: 1 & 2
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-5709
AUTHORITY: J. L. VOTA
ENGR. LTR. NO. EP/SA-36110

MICROFILM QUALITY ACCEPTABLE

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

DFTM. P HEARN
CHKR.
DES. ENG. 7/12/73
MFG. ENG.
MTLS. ENG.
APP.
APP.
APP.
DFTG. SUPV.

TITLE: GEORGIA POWER CO. ALVIN W VOGTLE UNITS 1 & 2 FUNCTIONAL DIAGRAM PRESSURIZER HEATER CONTROL

SCALE
DIMENSIONS IN INCHES
DO NOT SCALE
7243D07
SHEET - 12
SUB

NEXT ASSY.
REF. DWG.

1	2	3	4	5
GAE/GBE-300	GAE/GBE-300 ECN-31883	GAE/GBE-300 ECN-31242	GAE/GBE-300 ECN-31569	GAE/GBE-300 ECN-32037
	REVISIONS CIRCLED	REV'S CIRCLED	REV CIRCLED	CHANGES CIRCLED

Resub for micro film quality

DISTRIBUTION TO: FOR: REVIEW INFO
- MECHANICAL
  - BALANCE OF PLANT
  - BOILER/NSSS
  - PLANT UTILITIES
- PLANT DESIGN
- CONTROL SYSTEMS ✓
- ELECTRICAL ✓
  - WIRING
  - CONDUIT
- MOS
- PAINTING & COATINGS
- CIVIL/STRUCTURAL
- NUCLEAR
- STRESS
- ARCHITECTURAL
- STARTUP
- CONSTRUCTION
- NOT REQ'D BY ENGRG
- CLIENT

IDENTIFYING TITLE OF THIS DOCUMENT:

Bechtel Log No. 1 X6AA02-236-7

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DATE RECEIVED 4-9-82
DOCUMENT STATUS
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3 ☐ REVISE AND RESUBMIT. WORK MAY NOT PROCEED.
4 ☐ INFORMATION ONLY. DISTRIBUTION REQ'D ☐ YES
DATE 5-10-82

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DATE: 5-17-82 CAMERA OPERATOR SECTION SUPERVISOR

1. 028K1.01 001/LOIT/RO/C/A 2.5/2.5/028K1.01/LO-TA-29013///

Initial conditions:

- Unit 1 is at 100% reactor power.
- Containment Spray pump 'A' is tagged out.

Current conditions:

- Large break LOCA occurs.
- 19010-C, "Loss of Reactor or Secondary Coolant," is in progress.
- Containment Spray pump 'B' trips.
- Containment pressure is 22 psig.
- Containment hydrogen concentration is 5.1%.

Which one of the following completes the following statement?

The **preferred** method for reducing containment hydrogen concentration is __(1)__ containment atmosphere,

and

a __(2)__ path entry condition exists for 19251-C, "Response to High Containment Pressure."

	__(1)__	__(2)__
A.	diluting	red
B.✓	diluting	orange
C.	purging	red
D.	purging	orange

K/A

028 **Hydrogen Recombiner and Purge Control System**

K1.01 **Knowledge of the physical connections and/or cause-effect relationships between the HRPS and the following systems:**

- Containment annulus ventilation system (including pressure limits).

K/A MATCH ANALYSIS

The question presents the candidate with a plausible scenario in which conditions are provided in the stem that the candidate must use to determine the proper action based on his knowledge of the procedure steps and limitations, including specific alignments

allowed per the SOP. This also includes pressure limits allowed to perform the process and which method is preferred. In addition, the candidate must use the information provided to evaluate the challenge to the containment barrier and determine the proper procedural response.

EXPLANATION OF REQUIRED KNOWLEDGE

With containment hydrogen concentration at 5.1%, hydrogen concentratin must be reduced. Per SOP 13130-1, two methods are available to accomplish this task. The "preferred" method dilutes containment atmosphere with service air, and is the preferred method since it does not create an emergency release. However, this method is not to be performed if containment pressure is >40 psig, unless exceptional circumstances exist. The second method is also governed by SOP 13130-1 and requires Emergency Director approval prior to initiation because it results in an emergency exposure to the general public and the plant population.

Critical Safety Function Status Tree (CSFST) priority of operator action is based on the following colors in descending priority order: Red, Orange, Yellow, and Green. The Containment CSFST distinguishes between red and orange paths based on containment pressure above or below 52 psig. The distinction between orange and yellow paths is based on containment pressure above or below 21.5 psig, and whether a CSP is running. Based on containment pressure at 22 psig and neither CSP available, an Orange path is present per EOP 19200-C.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per SOP 13130-1, with containment pressure <40 psig, diluting containment with service air is the preferred method to prevent an emergency release.

The second part is incorrect. Per 19200-C, with containment pressure at 22 psig and no CS pumps available, an ORANGE path exists on the CSFST. However, if the candidate confused the first and second level of the containment pressure decision tree and believes the first decision is based on >21.5 psig, then a RED path would be correct.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per 19200-C, with containment pressure at 22 psig and no CS pumps available, an ORANGE path exists on the CSFST.

C. Incorrect. Plausible. The first part is incorrect. Per SOP 13130-1, with containment pressure <40 psig, diluting containment with service air is the preferred method to prevent an emergency release. However, a candidate without specific knowledge of the 40 psig limit may find it non-conservative to increase containment pressure with service air when it is currently >21.5 psig.

The second part is incorrect. See the second part of choice A

above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

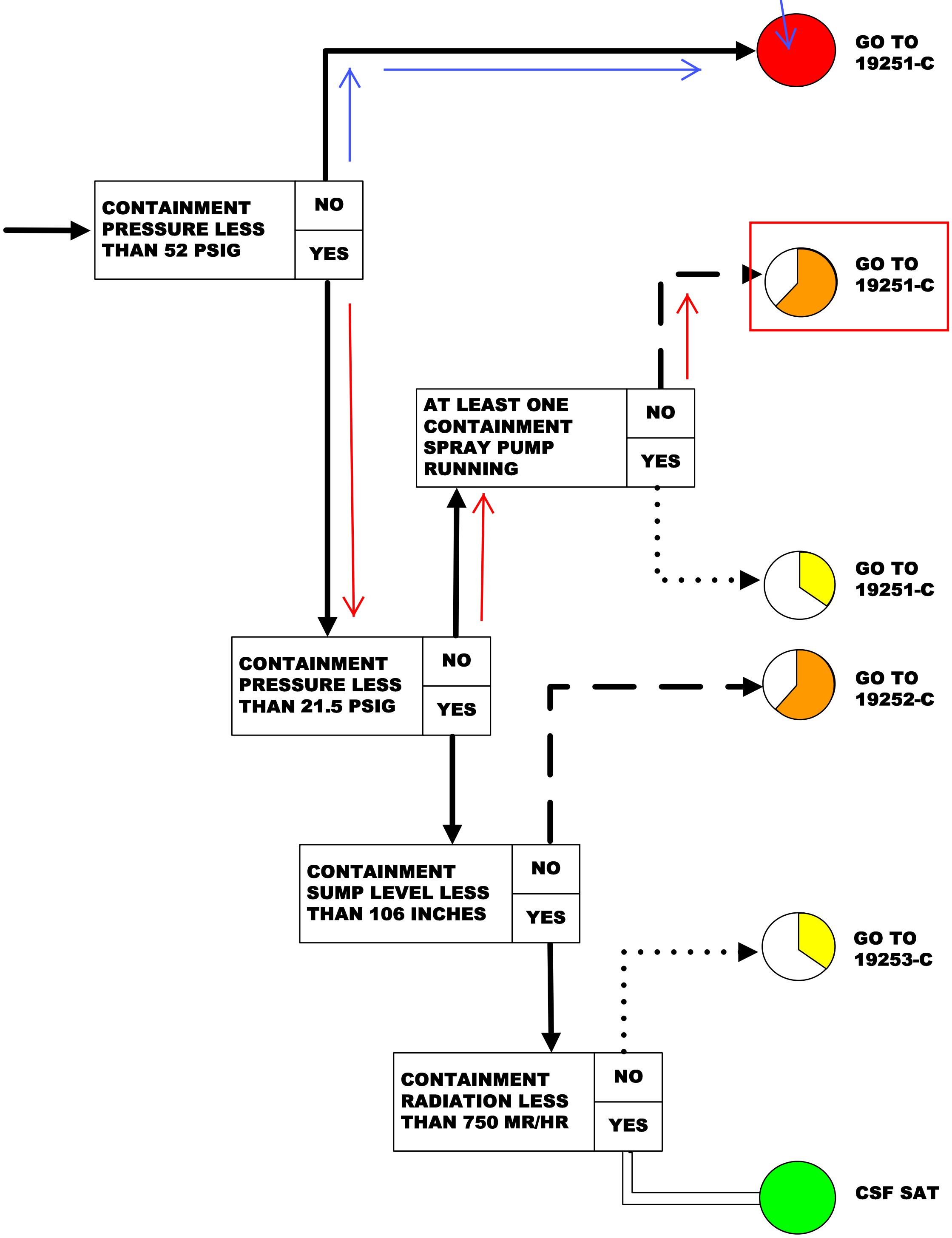
Level:	RO
Tier # / Group #	T2 / G2
K/A#	028K1.01
Importance Rating:	2.5 / 2.5
Technical Reference:	SOP 13130-1, Rev 20.0, pages 11 & 14 EOP 19200-C, Rev 24.2, page 9
References provided:	None
Learning Objective:	LO-LP-36107-03 State the means available to measure and control the containment hydrogen concentration LO-PP-29101-03 List the systems that are designed to control and mitigate hydrogen gas buildup in Containment. LO-PP-29101-18 State the upper and lower limits for an explosive mixture of hydrogen. LO-TA-29013 Perform a Dilution of Containment Hydrogen Concentration Using the Service Air System per 13130-1/2 LO-TA-29012 Operate the Post-LOCA Containment Hydrogen Purge system per 13130-1/2
Question origin:	NEW
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.2 to 41.9
Comments:	

You have completed the test!

Sheet 1 of 1

If candidate confuses the 52 psig tree with the 21.5 psig tree.

F- 0
CONTAINMENT



INITIALS

4.3 SHUTDOWN

NONE

4.4 NON PERIODIC OPERATION

4.4.1 Deleted

4.4.2 Diluting Containment Hydrogen Concentration Using The Service Air System

NOTES

Containment design pressure is 52 psig. ☐

CAUTION

Do not perform this section if containment pressure is greater than 40 psig unless so directed by the Emergency Director. ☐

4.4.2.1 **Reset** CIA by taking the following handswitches to RESET and **observe** ALB06-E06 extinguished:

- 1HS-40120 ________
- 1HS-40122 ________

4.4.2.2 **Open** SERVICE AIR CNMT HDR ISOL 1-HV-9385 as follows:

a. **Place** 1-HS-9385A on Main Control Room Panel QPCP to OPEN. ________

b. **Hold** 1-HS-9385B on Panel QPCP in OPEN until 1-HV-9385 is fully open. ________

INITIALS

4.4.3 Post-LOCA Containment Hydrogen Purge System Operation

CAUTIONS

- The Post-LOCA Containment Hydrogen Purge System is to be operated only if the containment hydrogen concentration cannot be maintained below 4% by other means. ☐
- The Post-LOCA Containment Hydrogen Purge System is designed to operate with a maximum pressure of 3 psi downstream of CNMT POST LOCA PURGE EXH DUCT CONTROL VLV 1-FV-2693. ☐
- Service air header pressure should be maintained greater than 80 psig to prevent header isolation while performing this section ☐

4.4.3.1 **Obtain** Emergency Director approval to perform this section: ________

4.4.3.2 **Verify** the Service Air System is operating. ________

4.4.3.3 **Place** disconnect switch at local Heater Control Panel 1-1508-N7-001-H01 to ON. ________

4.4.3.4 **Push** RESET button at local Heater Control Panel 1-1508-N7-001-H01 and **verify** that reset red light is ON. ________

Critical
4.4.3.5 Due to high radiation area potential, **verify** Containment Inside Isolation Valves 1-HV-2624A and 1-HV-2624B are closed and remain closed during the performance of the next step and until personnel have exited the area. ________

____
CV

4.4.3.6 **Unlock** and **open** POST LOCA PURGE CTB ISO VALVE 1-1508-U4-012. (KEY# 1OP3-381)[Equip. Bldg. roof (Dog House)] ________

4.4.3.7 **Reset** CVI by placing the following handswitches to the RESET position:

- 1HS-40121 ________
- 1HS-40123 ________

1. 029EG2.4.34 001/LOIT AND LOCT/RO/C/A 4.2/4.1/029EG2.4.34/LO-TA-37014///

Initial condition:

- Unit 1 is at 100% reactor power.

Current condition:

- ATWT is in progress.
- The reactor trip and bypass breakers will NOT open locally.

Which one of the following completes the following statement?

Per 19211-C, "Response to Nuclear Power Generation / ATWT," the Control Rod Drive MG Set __(1)__ breakers will be opened,

and

P-4 __(2)__ be generated when the MG set breakers are locally opened.

	__(1)__	__(2)__
A.	input	will
B.	input	will NOT
C.	output	will
D.✓	output	will NOT

K/A

029 ATWS

G2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the correct method to locally trip the reactor during an ATWS, and the resultant operational impact on turbine trip / reactor trip due to the P-4 signal not being generated because the reactor trip breakers remained closed.

EXPLANATION OF REQUIRED KNOWLEDGE

Per EOP 19211-C step 8, if the reactor has not tripped, the operator is directed to open the reactor trip and bypass breakers locally. If this action is unsuccessful, the operator is directed to open the MG set output breakers locally. The breakers can only be

operated locally and are immediately upstream of the RTBs. Opening these breakers will result in an immediate removal of power to the rod control cabinets and the reactor will trip. Conversely, if the MG set input breakers are opened, power is not immediately removed. Instead, the MG set will slowly coast down due to the large fly-wheel. As voltage decays, the stationary grippers will lose power and open, and the rods will fall. Since the voltage threshold can differ for each gripper, industry experience has shown that the rods will fall randomly and can result in significant local power peaking and fuel damage. (Reference V-LO-PP-27101 for a simplified diagram of Rod Control power.)

Since the reactor trip breakers are not open, a P-4 signal will not be generated. As a result, a turbine trip signal will not be generated on the reactor "trip" when the MG set output breakers are open. A manual turbine trip is required to ensure an excessive RCS cooldown does not occur when the reactor trips. (Reference 1X6AA02-00226 & 240)

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per EOP 19211-C step 8, the operator is directed to locally open the MG set output breakers. However, the MG set input breakers can be operated from the control room and are obviously labeled on the QEAB panel (see attached pictures of handswitches). Historically, prior to the industry OE, Vogtle's ATWS EOP directed opening the MG set input breakers.

The second part is incorrect. Reactor trip breakers remain closed resulting in neither train P-4 signal (which inserts the Turbine trip) being generated. The MG set breakers do not feed into the P-4 circuit. However, candidates normally think of a turbine trip resulting from a reactor trip. Since the reactor is tripped, it is reasonable for a candidate who does not know how P-4 is generated to assume P-4 would exist simply because the control cabinets are de-energzied.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. Reactor trip breakers remain closed resulting in neither train P-4 being generated. The MG set breakers do not feed into the P-4 circuit or directly cause a turbine trip.

C. Incorrect. Plausible. The first part is correct. Per EOP 19211-C step 8, the operator is directed to locally open the MG set output breakers.

The second part is incorrect. See the second part of choice A above.

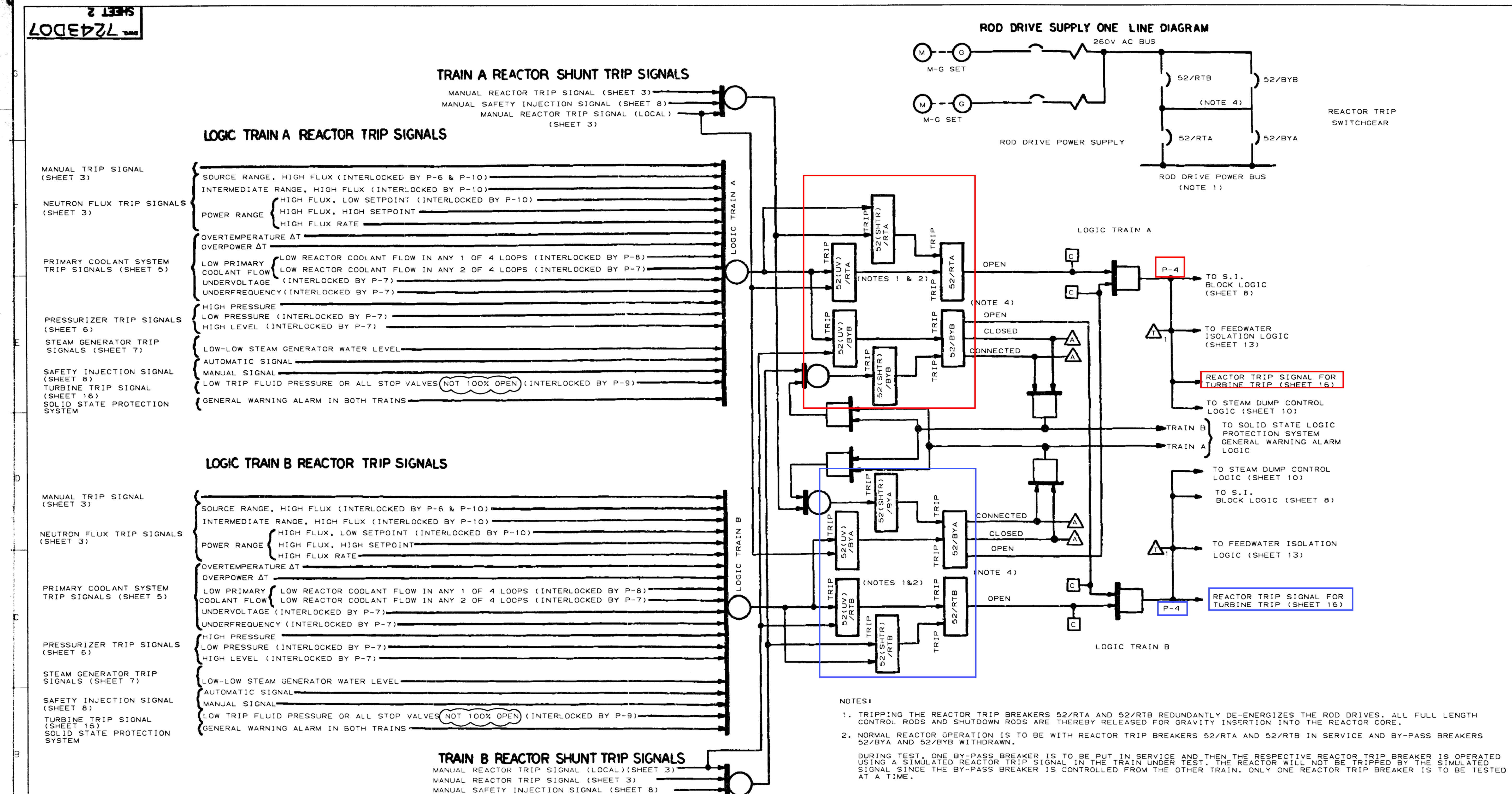
D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	029EG2.4.34
Importance Rating:	4.2 / 4.1
Technical Reference:	EOP 19211-C, Rev , page LOGIC 1X6AA02-00226, Rev 9.0 LOGIC 1X6AA02-00240, Rev 9.0 Lesson Plan V-LO-PP-27101, Rev 2.0, slide 32 Pictures of HS-1NB0801 and HS-1NB0901
References provided:	None
Learning Objective:	LO-PP-28103-02 List all permissives with applicable set points, coincidences, and functions. LO-PP-28103-03 List all reactor trip set points, coincidences, permissives, and blocks. LO-PP-27101-02 State the power supplies for the Rod Control System. LO-TA-37014 Respond to a Nuclear Power Generation/ ATWT Condition per 19211-C LO-TA-27008 Draw and label a one-line diagram of the Control Rod Drive Power Supply
Question origin:	NEW
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7 / 41.10
Comments:	

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***7. Check for SI:**	
__a. SI signal - EXISTS OR ACTUATED.	__a. IF an SI signal is actuated during this procedure, THEN initiate ATTACHMENT A. __ Go to Step 8
b. Initiate ATTACHMENT A.	
8. Check the following trips have occurred:	
__a. Reactor trip.	__a. Locally trip the Reactor trip and Bypass breakers. __ IF the trip breakers will NOT open, THEN trip the Control Rod Drive MG Set output breakers at the Reactor Trip Switchgear.
__b. Turbine trip.	__b. Dispatch operator to trip turbine at the HP Turbine front standard.
***9. Check Reactor power:**	
__a. LESS THAN 5%.	__a. Go to Step 10.
__b. IR SUR - LESS THAN 0 DPM.	__b. Go to Step 10.
__c. Go to Step 24.	



NOTES:

1. TRIPPING THE REACTOR TRIP BREAKERS 52/RTA AND 52/RTB REDUNDANTLY DE-ENERGIZES THE ROD DRIVES. ALL FULL LENGTH CONTROL RODS AND SHUTDOWN RODS ARE THEREBY RELEASED FOR GRAVITY INSERTION INTO THE REACTOR CORE.
2. NORMAL REACTOR OPERATION IS TO BE WITH REACTOR TRIP BREAKERS 52/RTA AND 52/RTB IN SERVICE AND BY-PASS BREAKERS 52/BYA AND 52/BYB WITHDRAWN.

   DURING TEST, ONE BY-PASS BREAKER IS TO BE PUT IN SERVICE AND THEN THE RESPECTIVE REACTOR TRIP BREAKER IS OPERATED USING A SIMULATED REACTOR TRIP SIGNAL IN THE TRAIN UNDER TEST. THE REACTOR WILL NOT BE TRIPPED BY THE SIMULATED SIGNAL SINCE THE BY-PASS BREAKER IS CONTROLLED FROM THE OTHER TRAIN. ONLY ONE REACTOR TRIP BREAKER IS TO BE TESTED AT A TIME.
3. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT BECAUSE BOTH TRAINS ARE SHOWN.
4. OPEN/CLOSED INDICATION FOR EACH TRIP BREAKER AND EACH BYPASS BREAKER IN CONTROL ROOM.

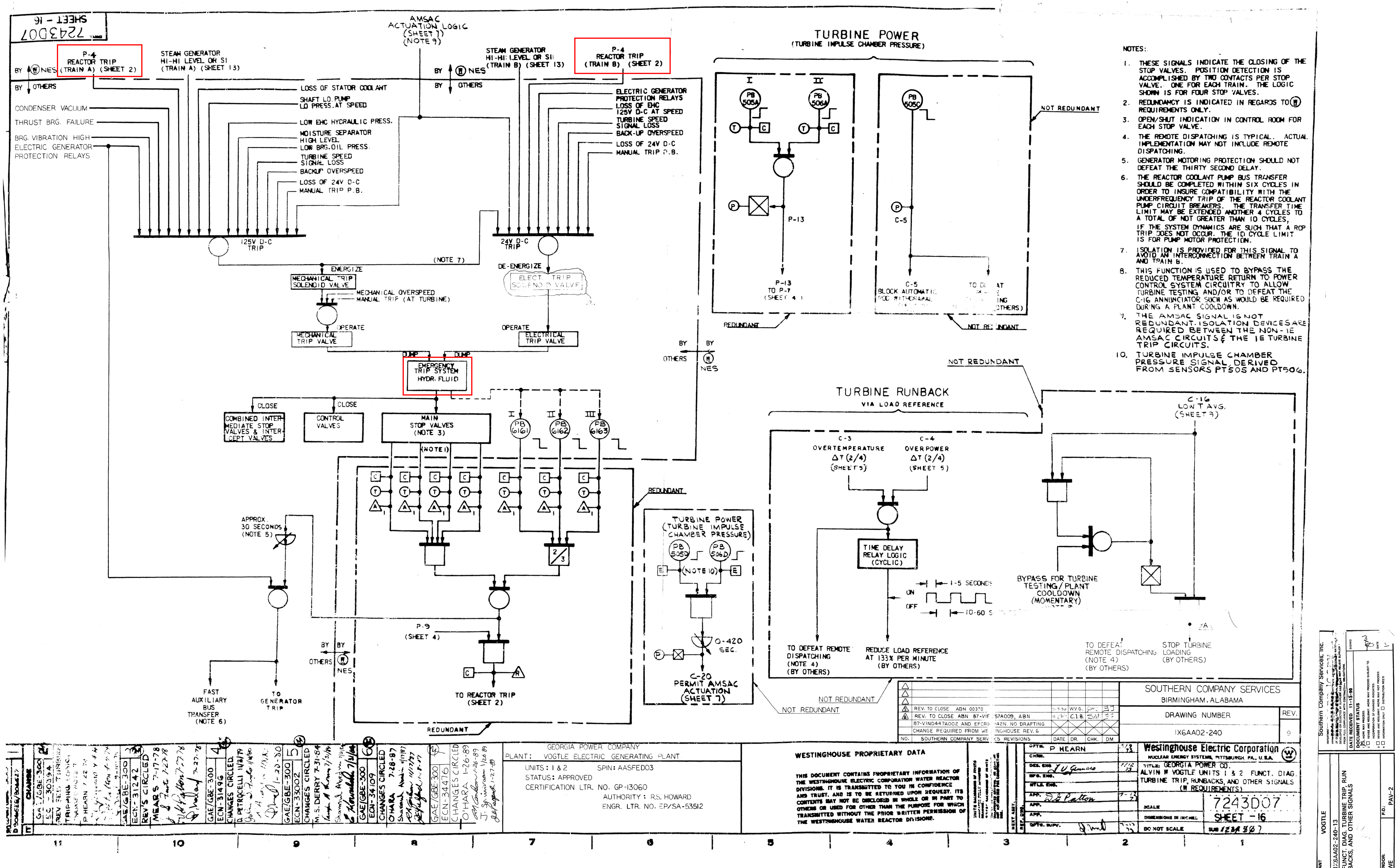
7 | REDRAWN BY SCS AND ISSUED TO INCORP. ABN-05114 | ABN-05114

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA U.S.A.
TITLE GEORGIA POWER CO
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
REACTOR TRIP SIGNALS
7243D07
SHEET 2
1X6AA02-226-9

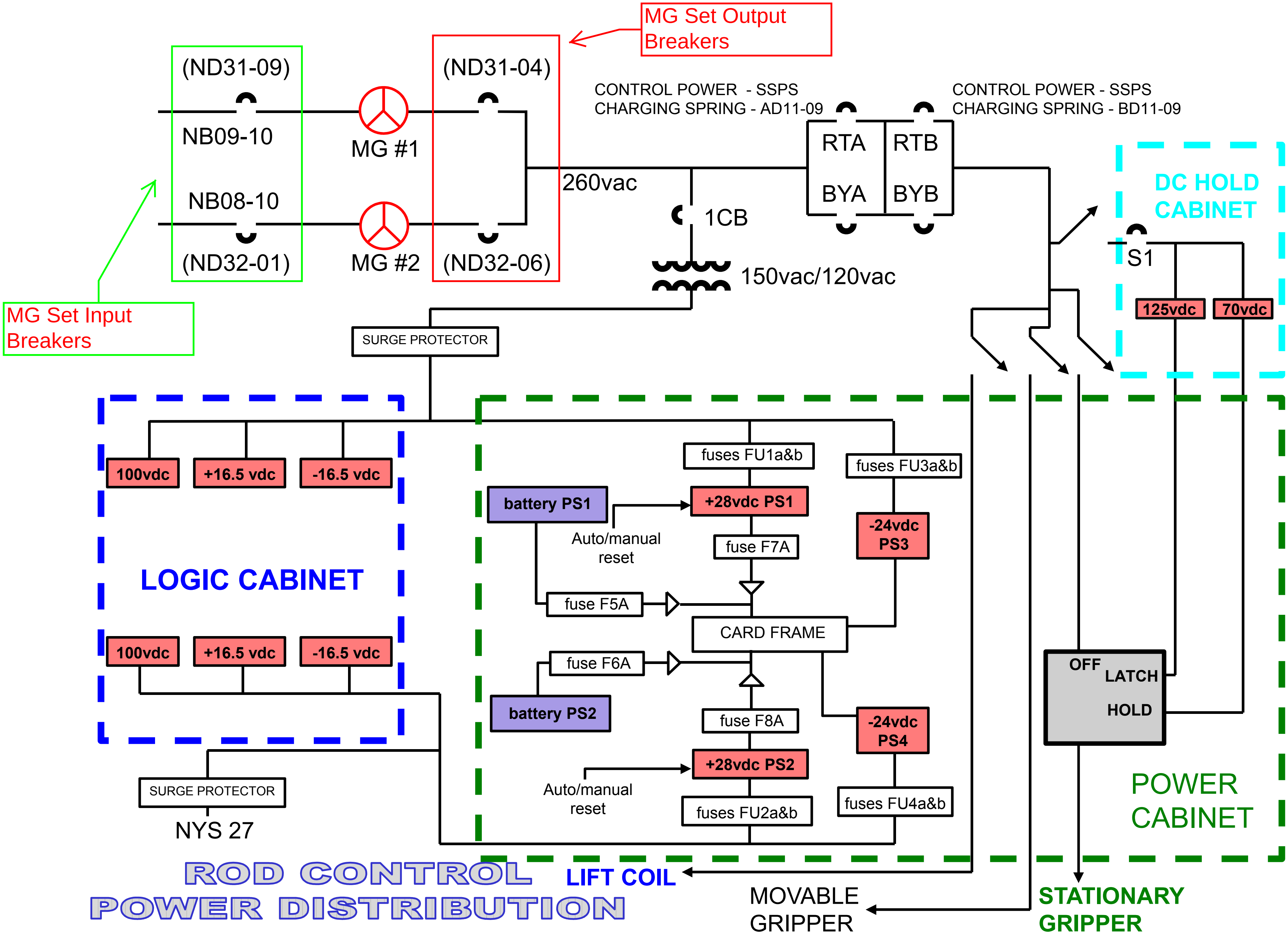
Southern Company Services, Inc.
DATE RECEIVED 12/03/93
DOCUMENT STATUS

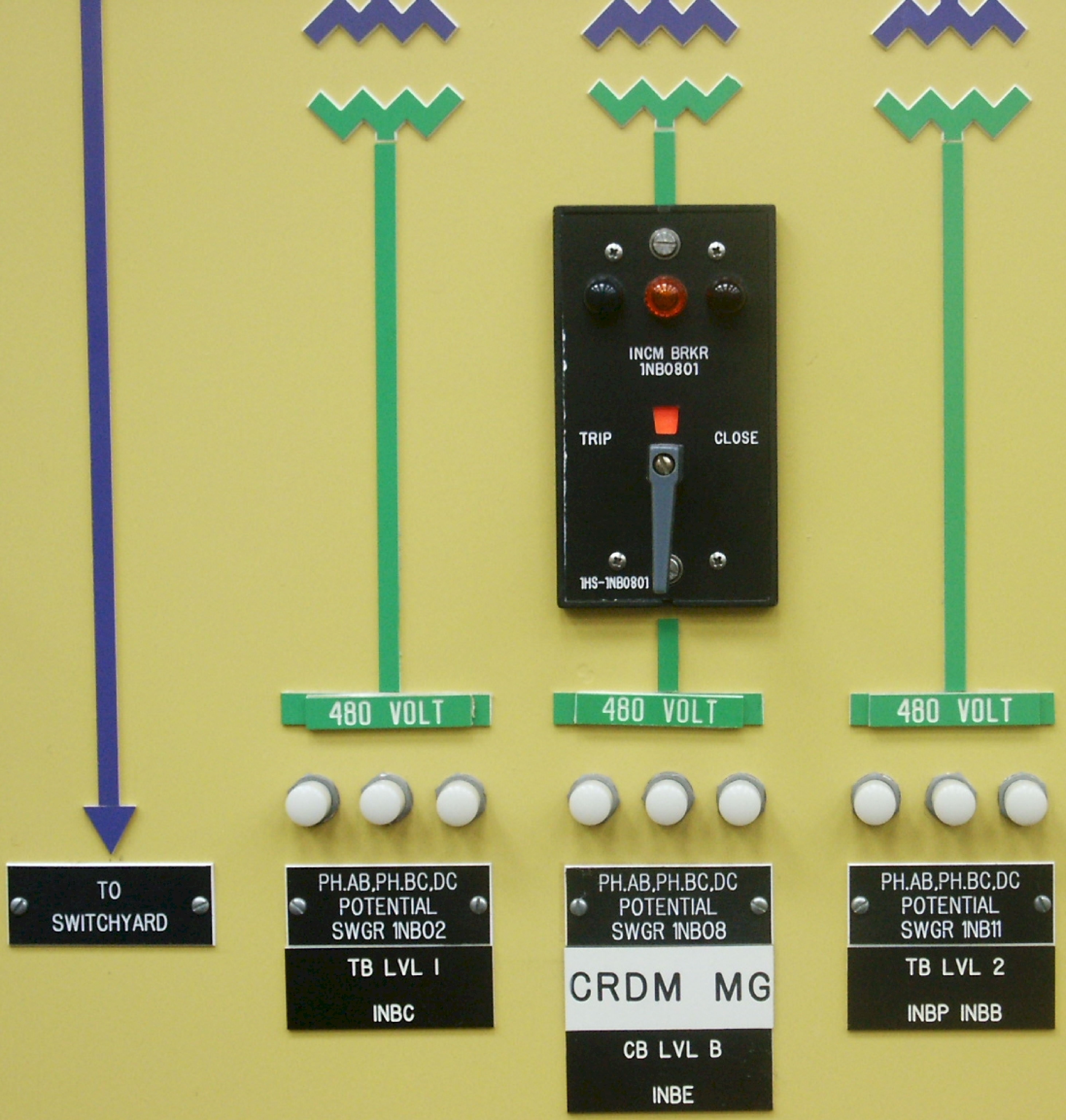
PLANT VOGTLE
1X6AA02-226-9
FUNCTIONAL DIAGRAMS REACTOR TRIP SIGNALS
(REDRAWN AND REVISED BY SCS TO INCORPORATE ABN-05114)
VENDOR: WE

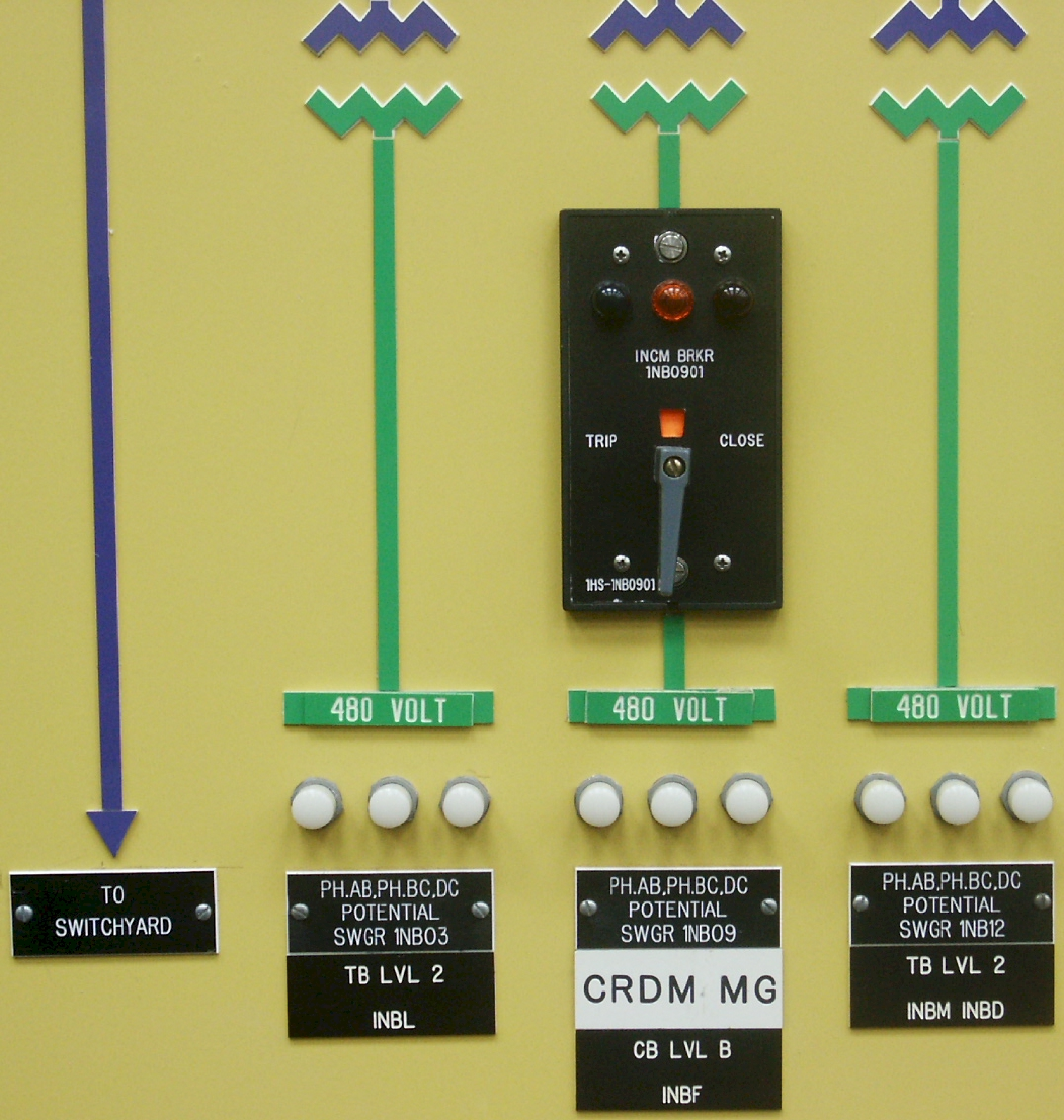
11 10 9 8 7 6 5 4 3 2 1



7243D07 SHEET - 16
AMSAC ACTUATION LOGIC (SHEET 7) (NOTE 9)
P-4 REACTOR TRIP (TRAIN A) (SHEET 2)
STEAM GENERATOR HI-HI LEVEL OR SI (TRAIN A) (SHEET 13)
STEAM GENERATOR HI-HI LEVEL OR SI (TRAIN B) (SHEET 13)
P-4 REACTOR TRIP (TRAIN B) (SHEET 2)
BY W NES
BY OTHERS
CONDENSER VACUUM
THRUST BRG. FAILURE
BRG. VIBRATION HIGH
ELECTRIC GENERATOR PROTECTION RELAYS
LOSS OF STATOR COOLANT
SHAFT LO PUMP LO PRESS. AT SPEED
LOW EHC HYDRAULIC PRESS.
MOISTURE SEPARATOR HIGH LEVEL
LOW BRG. OIL PRESS.
TURBINE SPEED SIGNAL LOSS
BACKUP OVERSPEED
LOSS OF 24V D-C
MANUAL TRIP P.B.
ELECTRIC GENERATOR PROTECTION RELAYS
LOSS OF EHC
125V D-C AT SPEED
TURBINE SPEED SIGNAL LOSS
BACK-UP OVERSPEED
LOSS OF 24V D-C
MANUAL TRIP P.B.
125V D-C TRIP
24V D-C TRIP
(NOTE 7)
ENERGIZE
DE-ENERGIZE
MECHANICAL TRIP SOLENOID VALVE
ELECT. TRIP SOLENOID VALVE
MECHANICAL OVERSPEED
MANUAL TRIP (AT TURBINE)
OPERATE
MECHANICAL TRIP VALVE
ELECTRICAL TRIP VALVE
DUMP
EMERGENCY TRIP SYSTEM HYDR. FLUID
CLOSE
COMBINED INTERMEDIATE STOP VALVES & INTERCEPT VALVES
CONTROL VALVES
MAIN STOP VALVES (NOTE 3)
(NOTE 1)
PB 6161
PB 6162
PB 6163
REDUNDANT
APPROX. 30 SECONDS (NOTE 5)
P-9 (SHEET 4)
FAST AUXILIARY BUS TRANSFER (NOTE 6)
TO GENERATOR TRIP
TO REACTOR TRIP (SHEET 2)
TURBINE POWER (TURBINE IMPULSE CHAMBER PRESSURE)
PB 505A
PB 506A
PB 505C
P-13
C-5
P-13 TO P-7 (SHEET 4)
C-5 BLOCK AUTOMATIC ROD WITHDRAWAL
NOT REDUNDANT
TURBINE POWER (TURBINE IMPULSE CHAMBER PRESSURE)
PB 505D
PB 506D
(NOTE 10)
0-420 SEC.
C-20 PERMIT AMSAC ACTUATION (SHEET 7)
TURBINE RUNBACK
VIA LOAD REFERENCE
C-3 OVERTEMPERATURE ΔT (2/4) (SHEET 5)
C-4 OVERPOWER ΔT (2/4) (SHEET 5)
C-16 LOW T AVG. (SHEET 9)
TIME DELAY RELAY LOGIC (CYCLIC)
1-5 SECONDS
ON
OFF
10-60 S
BYPASS FOR TURBINE TESTING / PLANT COOLDOWN (MOMENTARY)
TO DEFEAT REMOTE DISPATCHING (NOTE 4) (BY OTHERS)
REDUCE LOAD REFERENCE AT 133% PER MINUTE (BY OTHERS)
TO DEFEAT REMOTE DISPATCHING (NOTE 4) (BY OTHERS)
STOP TURBINE LOADING (BY OTHERS)
NOTES:
1. THESE SIGNALS INDICATE THE CLOSING OF THE STOP VALVES. POSITION DETECTION IS ACCOMPLISHED BY TWO CONTACTS PER STOP VALVE. ONE FOR EACH TRAIN. THE LOGIC SHOWN IS FOR FOUR STOP VALVES.
2. REDUNDANCY IS INDICATED IN REGARDS TO W REQUIREMENTS ONLY.
3. OPEN/SHUT INDICATION IN CONTROL ROOM FOR EACH STOP VALVE.
4. THE REMOTE DISPATCHING IS TYPICAL. ACTUAL IMPLEMENTATION MAY NOT INCLUDE REMOTE DISPATCHING.
5. GENERATOR MOTORING PROTECTION SHOULD NOT DEFEAT THE THIRTY SECOND DELAY.
6. THE REACTOR COOLANT PUMP BUS TRANSFER SHOULD BE COMPLETED WITHIN SIX CYCLES IN ORDER TO INSURE COMPATIBILITY WITH THE UNDERFREQUENCY TRIP OF THE REACTOR COOLANT PUMP CIRCUIT BREAKERS. THE TRANSFER TIME LIMIT MAY BE EXTENDED ANOTHER 4 CYCLES TO A TOTAL OF NOT GREATER THAN 10 CYCLES, IF THE SYSTEM DYNAMICS ARE SUCH THAT A RCP TRIP DOES NOT OCCUR. THE 10 CYCLE LIMIT IS FOR PUMP MOTOR PROTECTION.
7. ISOLATION IS PROVIDED FOR THIS SIGNAL TO AVOID AN INTERCONNECTION BETWEEN TRAIN A AND TRAIN B.
8. THIS FUNCTION IS USED TO BYPASS THE REDUCED TEMPERATURE RETURN TO POWER CONTROL SYSTEM CIRCUITRY TO ALLOW TURBINE TESTING AND/OR TO DEFEAT THE C-16 ANNUNCIATOR SUCH AS WOULD BE REQUIRED DURING A PLANT COOLDOWN.
9. THE AMSAC SIGNAL IS NOT REDUNDANT. ISOLATION DEVICES ARE REQUIRED BETWEEN THE NON-1E AMSAC CIRCUITS & THE 1E TURBINE TRIP CIRCUITS.
10. TURBINE IMPULSE CHAMBER PRESSURE SIGNAL DERIVED FROM SENSORS PT505 AND PT506.
SOUTHERN COMPANY SERVICES
BIRMINGHAM, ALABAMA
DRAWING NUMBER 1X6AA02-240
REV. 9
REV. TO CLOSE ABN 87-VIF 57A009, ABN
87-VIN0447A002 AND EFCR
CHANGE REQUIRED FROM WESTINGHOUSE REV. 6
SOUTHERN COMPANY SERVICES REVISIONS
GEORGIA POWER COMPANY
PLANT: VOGTLE ELECTRIC GENERATING PLANT
UNITS: 1 & 2
SPIN: AASFED03
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-13060
AUTHORITY: R.S. HOWARD
ENGR. LTR. NO. EP/SA-53512
WESTINGHOUSE PROPRIETARY DATA
THIS DOCUMENT CONTAINS PROPRIETARY INFORMATION OF THE WESTINGHOUSE ELECTRIC CORPORATION WATER REACTOR DIVISIONS. IT IS TRANSMITTED TO YOU IN CONFIDENCE AND TRUST, AND IS TO BE RETURNED UPON REQUEST. ITS CONTENTS MAY NOT BE DISCLOSED IN WHOLE OR IN PART TO OTHERS OR USED FOR OTHER THAN THE PURPOSE FOR WHICH TRANSMITTED WITHOUT THE PRIOR WRITTEN PERMISSION OF THE WESTINGHOUSE WATER REACTOR DIVISIONS.
Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.
P HEARN
GEORGIA POWER CO.
ALVIN W VOGTLE UNITS 1 & 2 FUNCT. DIAG.
TURBINE TRIP, RUNBACKS, AND OTHER SIGNALS
(W REQUIREMENTS)
7243D07
SHEET - 16
DO NOT SCALE
VOGTLE
1X6AA02-240-13
FUNCT. DIAG. TURBINE TRIP, RUN BACKS, AND OTHER SIGNALS
PAV-2
WE
Southern Company Services, Inc.


MG Set Output Breakers
MG Set Input Breakers
(ND31-09)
NB09-10
NB08-10
(ND32-01)
MG #1
MG #2
(ND31-04)
(ND32-06)
260vac
1CB
150vac/120vac
CONTROL POWER - SSPS CHARGING SPRING - AD11-09
CONTROL POWER - SSPS CHARGING SPRING - BD11-09
RTA
RTB
BYA
BYB
DC HOLD CABINET
S1
125vdc
70vdc
SURGE PROTECTOR
LOGIC CABINET
100vdc
+16.5 vdc
-16.5 vdc
100vdc
+16.5 vdc
-16.5 vdc
SURGE PROTECTOR
NYS 27
fuses FU1a&b
fuses FU3a&b
battery PS1
+28vdc PS1
-24vdc PS3
Auto/manual reset
fuse F7A
fuse F5A
CARD FRAME
fuse F6A
battery PS2
fuse F8A
-24vdc PS4
+28vdc PS2
Auto/manual reset
fuses FU2a&b
fuses FU4a&b
OFF
LATCH
HOLD
POWER CABINET
ROD CONTROL POWER DISTRIBUTION
LIFT COIL
MOVABLE GRIPPER
STATIONARY GRIPPER


INCM BRKR
1NB0801
TRIP
CLOSE
1HS-1NB0801
480 VOLT
480 VOLT
480 VOLT
TO
SWITCHYARD
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB02
TB LVL I
INBC
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB08
CRDM MG
CB LVL B
INBE
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB11
TB LVL 2
INBP INBB


INCM BRKR
1NB0901
TRIP
CLOSE
1HS-1NB0901
480 VOLT
480 VOLT
480 VOLT
TO
SWITCHYARD
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB03
TB LVL 2
INBL
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB09
CRDM MG
CB LVL B
INBF
PH.AB,PH.BC,DC
POTENTIAL
SWGR 1NB12
TB LVL 2
INBM INBD

1. 033AG2.2.44 001/LOIT/RO/C/A 4.2/4.4/033AG2.2.44/LO-TA-60035///

Initial conditions:

- Unit 1 reactor startup is in progress.
- Critical data is being collected per 12003-C, "Reactor Startup (Mode 3 to Mode 2)."

Current condition:

- Intermediate Range N35 fails **bottom** of scale.

Which one of the following completes the following statement?

Based on the current conditions, the SR BLOCK PERMISSIVE P-6 light on the BPLB __(1)__ remain lit,

and

placing the affected channel's Level Trip switch in BYPASS per 18002-C, "Nuclear Instrumentation System Malfunction," aligns NI __(2)__ power to the SSPS input relay.

	__(1)__	__(2)__
A.✓	will	control
B.	will	instrument
C.	will NOT	control
D.	will NOT	instrument

K/A

033 Loss of Intermediate Range NI

G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A MATCH ANALYSIS

The question has the candidate determine the status of P-6 on the BPLB based on the plant conditions. The question then requires the candidate to demonstrate understanding of the effect of placing the NI channel in BYPASS during the recovery actions.

EXPLANATION OF REQUIRED KNOWLEDGE

At the point of taking critical data, reactor power is 2 x 10-3. P-6 permissive has been

met and SR Hi Flux trips have been blocked. When N35 fails low, one of two P-6 bistables de-energizes. However, one of two bistables remains and maintains the P-6 permissive. Therefore, the P-6 BPLB remains lit.

Per 18002-C direction, the operating crew will place the affected channel's LEVEL TRIP switch to the BYPASS position. This aligns control power directly to the SSPS input bay contacts, preventing the NI channel from causing a tripped input and allowing I&C to troubleshoot the instrument.

There are two fuses on the NI drawers - control and instrument. The control power fuse will de-energize the instrument power as well as all the bistables in the NI drawer and allow SSPS to sense a tripped condition. The instrument power fuse will cause the instrument output to fail downscale low and the NI bistables to trip. However, the input bay of SSPS is unchanged.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. When N35 fails low, one of two P-6 bistables de-energizes. However, one of two bistables remains and maintains the P-6 permissive. Therefore, the P-6 BPLB remains lit.

The second part is correct. The control power fuse will de-energize the instrument power as well as all the bistables in the NI drawer and allow SSPS to sense a tripped condition.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. Control power feeds the bistables in the NI drawers. However, there are two power sources to the NI drawers - control and instrument. These power sources are frequently confused by intial candidates as to how they function.

C. Incorrect. Plausible. The first part is incorrect. When one of two IR detectors reaches 2 x10-5, the P-6 BPLB illuminates. Losing one of two IR detectors leaves one of two IR detectors available to meet the permissive. Therefore, the bistable does not change state. On lowering power two of two TSLBs for P-6 must extinguish for the P-6 BPLB to extinguish and automatically unblock the SR Hi Flux Trips. Candidates frequently confuse the order and coincidence of these light indications.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G2
K/A#	033AG2.2.44
Importance Rating:	4.2 / 4.4

Technical Reference: 1X6AA02-00227 Rev 9.0
1X6AA02-00227 Rev 8.0
18002-C Rev 20.1
V-LO-PP-17201 Rev 3.0

References provided: None

Learning Objective:

V-LO-PP-17201-01 Discuss the operation of the Source & Intermediate Range Detectors to include:

d. All Reactor Trip signals
e. All Permissives & Interlocks
g. Power supplies (also including the effects on loss of instrument or control power)

LO-TA-60035 Respond to Nuclear Instrumentation System Malfunction per 18002-C

LO-TA-61001 Reactor Startup using 12003-C

LO-TA-61003 Reactor Shutdown to Hot Standby using 12005-C

Question origin: MODIFIED - LOIT Question 033AA2.03-1

Cognitive Level: C/A

10 CFR Part 55 Content: 41.5 / 43.5 / 45.12

Comments:

You have completed the test!

1. 033AA2.03 001/1/2/LOSS OF IRNI/C/A-3.1/NEW/SRO/HL-15R AUDIT/DNS/

(SRO ONLY)

Original Question

Initial conditions:

- The unit is at 35% power.
- All equipment is aligned for automatic control.

Current conditions:

- SR & IR NI channels N31 & N35 level meters drop to zero.
- SUR indications for SR & IR SUR channels N31 & N35 remain on zero.
- TSLB Channel 1 SR & IR high flux trip bistables illuminate.
- TSLB Channel 1 P6 bistable goes dark.
- All BPLB lights remain unchanged.

Which one of the following describes the correct diagnosis for the 'A' Train SR/IR Control Room Signal Processor, and the actions to take per 18002-C, Section A, "Source / Intermediate Range Channel Malfunction"?

A. Loss of instrument power.

Restore inoperable Intermediate Range Channel prior to raising power. (TS 3.3.1)

B. Loss of control power.

Restore inoperable Intermediate Range Channel prior to raising power. (TS 3.3.1)

C.✓ Loss of instrument power.

Place LEVEL TRIP switch in BYPASS on the affected Source/Intermediate Range Drawer.

D. Loss of control power.

Place LEVEL TRIP switch in BYPASS on the affected Source/Intermediate Range Drawer.

A. SOURCE / INTERMEDIATE RANGE CHANNEL MALFUNCTION

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION

Two redundant drawer assemblies should not be placed in test at the same time.

A27. Perform the following on the affected Source/Intermediate Range Drawer:

a. Place LEVEL TRIP switch in BYPASS.

b. Check LEVEL TRIP BYPASS light – ON.

c. Block SR High Flux At Shutdown alarm.

d. Select audio count rate to operating channel, if required.

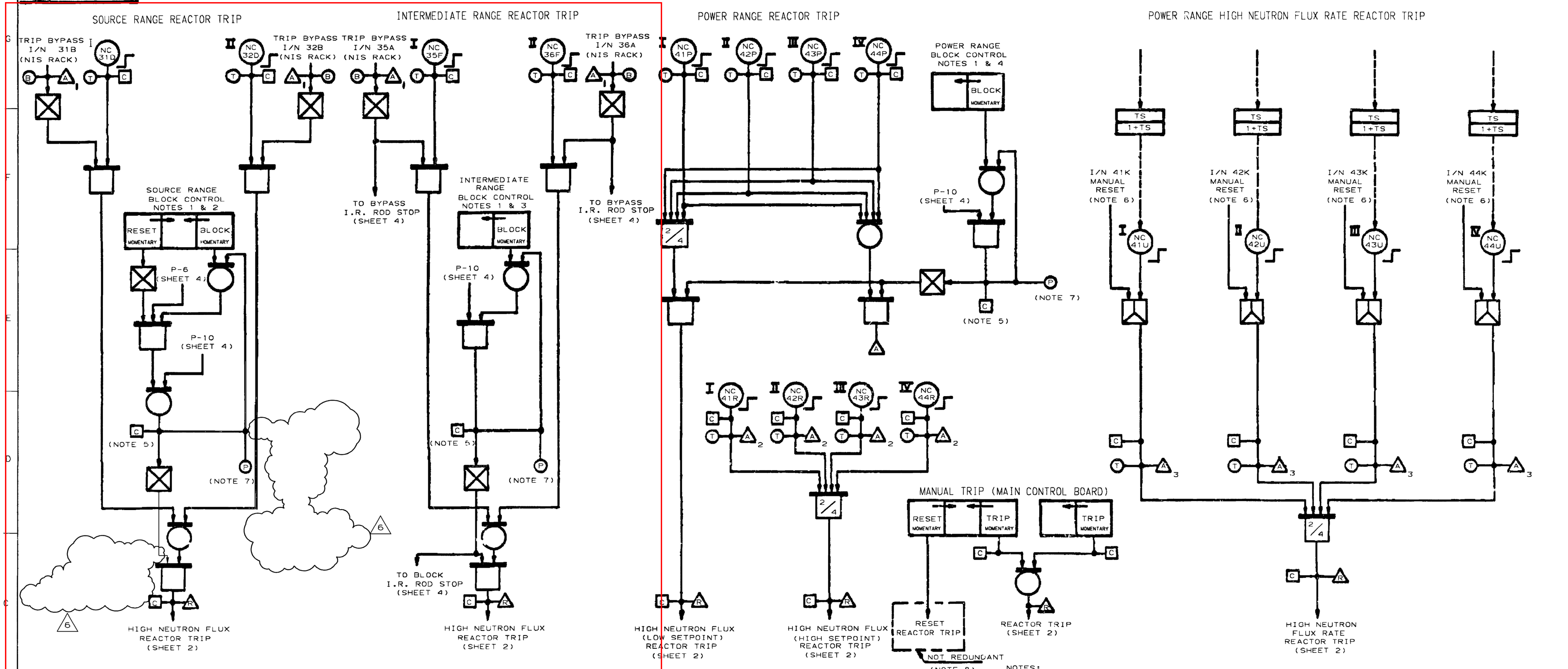
e. Select startup rate to operating channel, if appropriate.

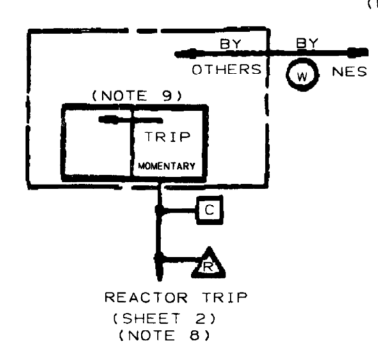
A28. Notify I&C to initiate repairs.

A29. Bypass the affected channels by initiating 13509-C, BYPASS TEST INSTRUMENTATION (BTI) PANEL OPERATION, if desired.

A30. Initiate 13501, NUCLEAR INSTRUMENTATION SYSTEM when repairs are complete.

7243D07 SHEET - 3





NOTES:

1. THE REDUNDANT MANUAL BLOCK CONTROLS CONSIST OF TWO CONTROLS ON THE CONTROL BOARD FOR EACH RANGE. ONE FOR EACH TRAIN.
2. I/N 33A IS IN LOGIC TRAIN A.
   I/N 33B IS IN LOGIC TRAIN B.
3. I/N 38A IS IN LOGIC TRAIN A.
   I/N 38A IS IN LOGIC TRAIN B.
4. I/N 47A IS IN LOGIC TRAIN A
   I/N 47B IS IN LOGIC TRAIN B.
5. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT. INDIVIDUAL FOR EACH TRAIN.
6. MANUAL RESET CONTROLS CONSIST OF FOUR MOMENTARY CONTROLS IN THE CONTROL ROOM. ONE CONTROL FOR EACH INSTRUMENT CHANNEL.
7. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCUIT. INDIVIDUAL FOR EACH TRAIN.
8. THIS SIGNAL GOES TO THE REACTOR TRIP BREAKERS.
9. THE REDUNDANT MANUAL REACTOR TRIP CONSISTS OF TWO CONTROLS ON THE LOCAL SHUTDOWN PANEL. ONE FOR EACH TRAIN. SUPPLIED BY OTHERS.

6	INCORPORATED PER DCP 96-V1N0044	05/12/99	ELC	EOG	GLB	
NO.	REVISIONS	DATE	DR	CHK	APPV	

SCS REVISIONS

5 — GBE-300 — ECN-34937 — CHANGES CIRCLED REDRAWN AND RE-ISSUED BY SCS TO INCORP. ABN-11479 — GAE

DOCUMENT STATUS CODE 1

1X6AA02-00227-9

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURG, PA. U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
NUCLEAR INSTR. & MANUAL TRIP SIGNALS

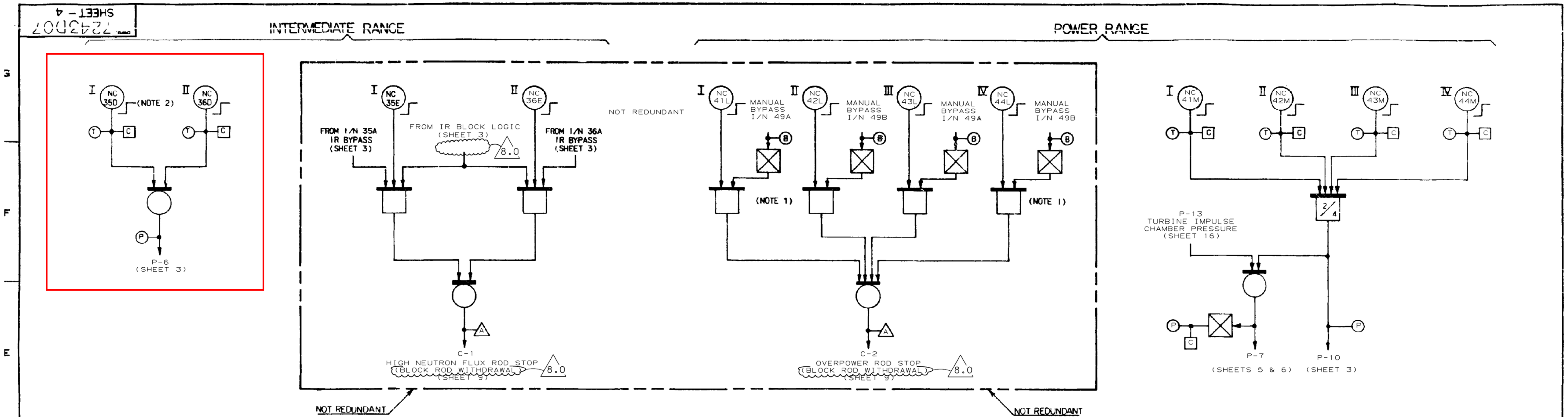
P. HEARN

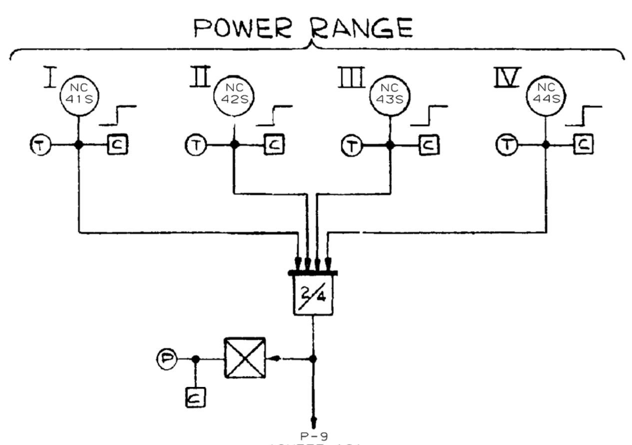
7243D07 SHEET 3

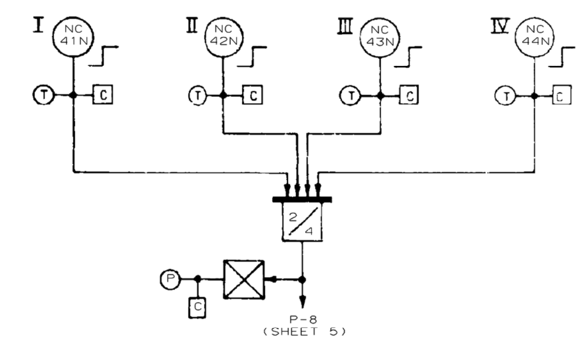
REV 6

SIZE "D" 1

CAD NAME 16A02227







NOTES:

1. THE BYPASS SIGNALS ARE MADE UP BY MEANS OF TWO THREE-POSITION SWITCHES ON A NIS RACK. SWITCH I/N 49A BYPASSES EITHER NC-41L OR NC-43L. SWITCH I/N 49B BYPASSES EITHER NC-42L OR NC-44L.

2. THE TWO P-6 BISTABLES NO. NC-35D AND NC-36D ARE "ENERGIZED TO ACTUATE" SUCH THAT A LOGIC 1 SIGNAL IS DEFINED TO BE PRESENT WHEN THE BISTABLE OUTPUT VOLTAGE IS ON.

8.0

SO	SAE/SBE-300
D	906528/906527
1	CHANGE
2	SAE/GBE-300 ECN-31242 REV'S CIRCLED YEARS 7-13-8

DFTM.	P HEARN
CHKD.	
DES. ENG.	J U Glenn
MFG. ENG.	
MTLS. ENG.	
APP.	Patton
APP.	
APP.	
DFTG. SUPV.	

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAMS
NUCLEAR INSTR. PERMISSIVES & BLOCKS

SCALE — DIMENSIONS IN INCHES — DO NOT SCALE

7243D07
SHEET - 4
SUB 12

THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
TAB/SECT. N/A
PAGE N/A
FIGURE N/A

VERSION 8.0 | DATE 11/04/09

REVISED BY SNC PER ABN-V01192, VER. 2.0.

SEE MICROFILM FOR PREVIOUS REV. SIGNATURES.

BY	CHK'D	APPR.1	APPR.2
CYN	RKF	PML	

CAD MICROSTATION CYN 02

1X6AA02-00228.DGN

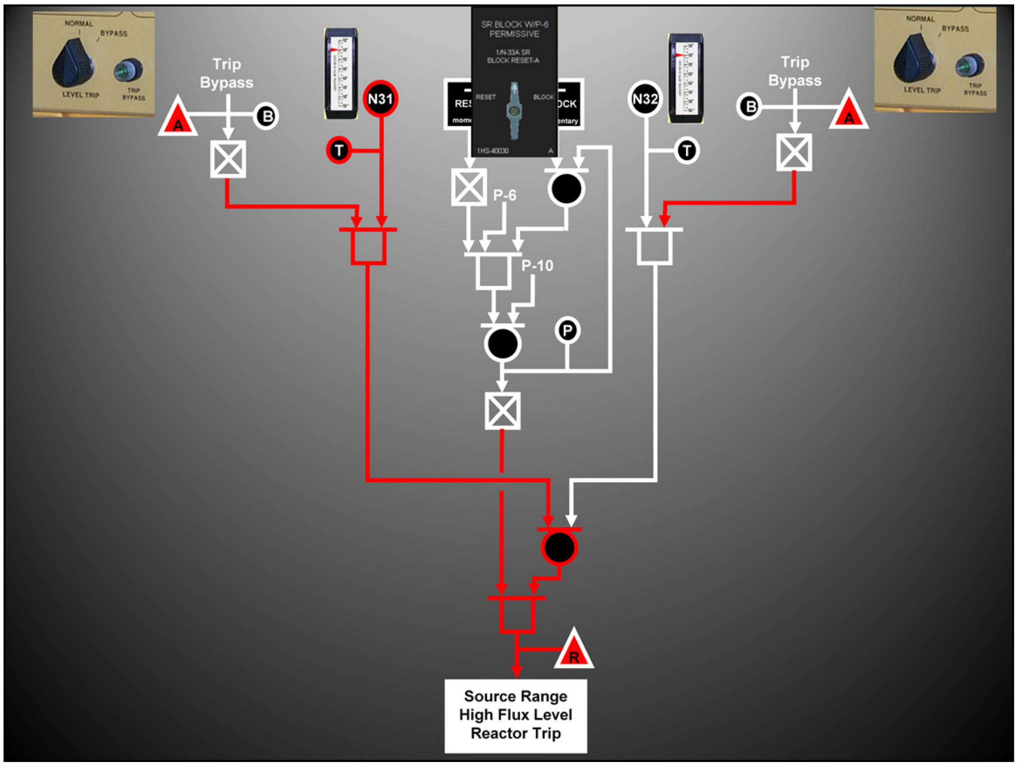
Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT
UNIT NO. 1

TITLE:
FUNCTIONAL DIAGRAMS
NUCLEAR INSTRUMENT PERMISSIVES & BLOCKS

VENDOR: WESTINGHOUSE | P.O.#:

DRAWING NO. 1X6AA02-00228

SIZE D

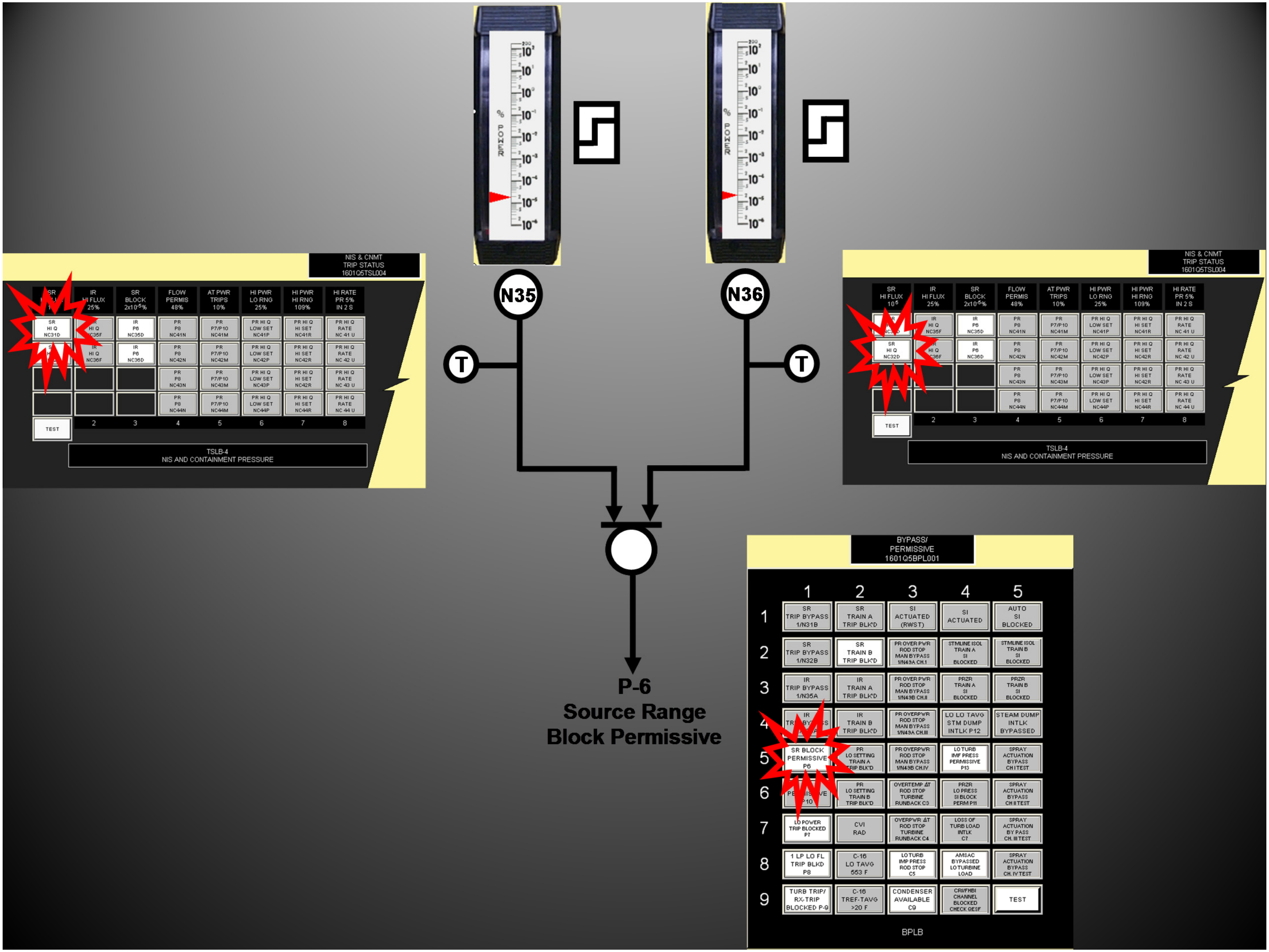


If either channel fails HIGH or exceeds 10^5 cps, a reactor trip signal will be generated unless it has been manually blocked above P-6 or power is above the P-10 setpoint.

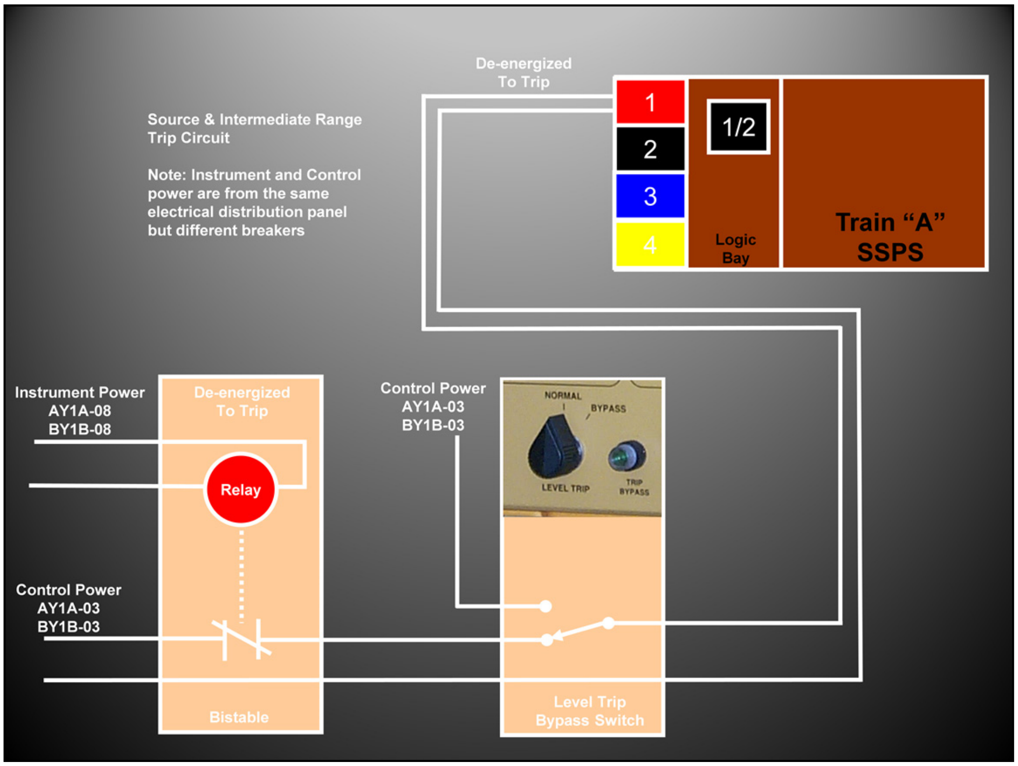
The manual block allows operation of the plant during startup conditions above the level trip set point. This is done after the Intermediate Range Nuclear Instruments have confirmed operation by coming on-scale at the proper flux level and have reached a preset **Permissive** set point **(1/2 Intermediate Range Channels $\geq$ 2x10^{-5} % Reactor Power).** This is also called **"P-6".**

(Objective V-LO-PP-17201-01d)

(Objective V-LO-PP-17201-01e)



N35
N36
T
T
NIS & CNMT
TRIP STATUS
1601Q5TSL004
TSLB-4
NIS AND CONTAINMENT PRESSURE
BYPASS/
PERMISSIVE
1601Q5BPL001
SR TRIP BYPASS 1/N31B
SR TRAIN A TRIP BLK'D
SI ACTUATED (RWST)
SI ACTUATED
AUTO SI BLOCKED
SR TRIP BYPASS 1/N32B
SR TRAIN B TRIP BLK'D
IR TRIP BYPASS 1/N35A
IR TRAIN A TRIP BLK'D
IR TRAIN B TRIP BLK'D
LO LO TAVG STM DUMP INTLK P12
STEAM DUMP INTLK BYPASSED
SR BLOCK PERMISSIVE P6
LO POWER TRIP BLOCKED P7
CVI RAD
1 LP LO FL TRIP BLKD P8
C-16 LO TAVG 553 F
TURB TRIP/ RX-TRIP BLOCKED P-9
C-16 TREF-TAVG >20 F
CONDENSER AVAILABLE C9
TEST
BPLB
P-6
Source Range
Block Permissive



The most important point to get from this slide is that in BYPASS, the input to SSPS is energized from control power (instrument power ONLY supplies indication when the switch is in bypass.

Questions (assume power is < P-10):

What happens on a loss of instrument power with LEVEL TRIP in NORMAL?

-Channel indication is LOST and the reactor TRIPS.

What happens on a loss of instrument power with LEVEL TRIP in BYPASS?

-Channel indication is LOST and the reactor does NOT trip.

What happens on a loss of control power?

-Channel indication is AVAILABLE and the reactor TRIPS.

(OBJECTIVE V-LO-PP-17201-01g)

Remember that there is only one set of breakers for both SR and IR, and the effect of the loss of power to BOTH instruments must be considered. Also, the status of P-6, P-10, and whether SR or IR trips has been blocked may be very important to properly answering the question.

1. 034A4.01 001/LOIT/RO/M/F 3.3/3.7/034A4.01/LO-PP-32101-08///

Given the following:

- Unit 1 is in Mode 6.
- Preparation for fuel movement is in progess in the Fuel Handling Building.

- The following radiation monitors are in service:

ARE-2532A/B, Fuel Handling Building Effluent Radiogas Monitors
1RE-008, Fuel Handling Building Area Monitor

Which one of the following completes the following statement?

ARE-2532A/B __(1)__ be monitored and the alarm setpoints adjusted from the SRDC panel,

and

1RE-008 __(2)__ be monitored and the alarm setpoints adjusted from the SRDC panel.

	__(1)__	__(2)__
A.	can	can
B.✓	can	can NOT
C.	can NOT	can
D.	can NOT	can NOT

K/A

034 Fuel Handling Equipment

A4.01 Ability to manually operate and/or monitor in the control room:

- Radiation levels.

K/A MATCH ANALYSIS

The KA addresses the relationship between fuel handling activities and the ability to monitor radiation levels and make required setpoint adjustments. The question has all the required elements to include the activity taking place in the fuel handling building and where the control room staff can monitor and adjust the setpoints for the radiation monitors.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17102-1, rad monitor A-RE-2532A/B is located on the SRDC in the control

room, where it can be monitored and its setpoints changed. 1RE-0008 can be monitored from the control room on either the IPC or Perms Console. However, since it is not located on the SRDC, its setpoint cannot be changed from the control room.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. A-RE-2532A/B is one of the radiation monitors that is operated from the main control room SRDC panel.

The second part is incorrect. 1RE-0008 is not one of the radiation monitors that is operated from the main control room SRDC panel. However, ARE-2532A/B and 1RE-008 are both listed in ARP 17100-C. The ARE-2532A/B entry sends you to ARP 17102-1 for the actions. A candidate without specific knowledge of rad monitor locations may remember seeing both listed and assume that both are on the SRDC.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. RE-0008 is not one of the radiation monitors that located on the main control room SRDC panel.

C. Incorrect. Plausible. The first part is incorrect. A-RE-2532A/B is one of the radiation monitors that is operated from the main control room SRDC panel. However, a candidate without specific knowledge of rad monitor locations may assume that since this is an "A" or common rad monitor, it would not be located on the Unit 1 SRDC panel.

The second part is incorrect. 1RE-0008 is not one of the radiation monitors that is operated from the main control room SRDC panel. However, if the candidate used the logic described in the first part of choice C above, it could be reasonable to assume 1RE-008 is located on the Unit 1 SRDC.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	034A4.01
Importance Rating:	3.3 / 3.7

Technical Reference: ARP 17100-1, Rev 26.2, pages 6-8
ARP 17102-1, Rev 20.2, page 4

References provided: None

Learning Objective: LO-PP-32101-08 List all safety-related radiation monitors by tag number and name. Describe those automatic actions that occur for each of the following safety-related monitors when its high alarm setpoint is exceeded:
b. fuel handling building effluent (ARE-2532A, B and ARE 2533A, B)

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.13

Comments:

You have completed the test!

ANNUNCIATOR RESPONSE INDEX

RADIATION MONITOR INDEX

* Safety Related. Go to 17102-1, "ARP For The SRDC QRM2".
** Not Functional - Detectors Removed.
*** Passive collector. No electronic components.

ANNUNCIATOR RESPONSE INDEX

RADIATION MONITOR INDEX

* Safety Related. Go to 17102-1, "ARP For The SRDC QRM2".
** Not Functional - Detectors Removed.
*** Passive collector. No electronic components.

ANNUNCIATOR RESPONSE INDEX

RADIATION MONITOR INDEX

* Safety Related. Go to 17102-1, "ARP For The SRDC QRM2".
** Not Functional - Detectors Removed.
*** Passive collector. No electronic components.

SAFETY RELATED DISPLAY CONSOLE

A1	
A2	
A3	
A4	
A5	1-RE-2562A
A6	1-RE-2562C
A7	
A8	
B1	1-RE-13119
B2	1-RE-13120
B3	1-RE-0002
B4	1-RE-0005
B5	1-RE-12116
B6	A-RE-2532A
B7	A-RE-2532B
B8	
C1	1-RE-13121
C2	1-RE-13122
C3	1-RE-0003
C4	1-RE-0006
C5	1-RE-12117
C6	A-43-2533A
C7	A-RE-2533B
C8	
D1	
D2	
D3	
D4	
D5	
D6	
D7	
D8	

SRDC CHANNEL DISPLAYS

1. 038EA2.05 001/LOIT/RO/2.8/2.9/038EA2.05/LO-TA-37011///

Given the following:

- Unit 1 experienced a SGTR.
- 19030-C, "Steam Generator Tube Rupture," is in progress.
- 1NAA is de-energized due to a bus fault.
- Crew is preparing to perform a maximum rate cool down.

Which one of the following completes the following statement?

In accordance with 19030-C, intact SG AFW flow rates are __(1)__,

and

then the cooldown is initiated per Step 17 using __(2)__.

A. (1) raised to prevent AFW re-initiation

(2) intact SG ARVs

B.✓ (1) raised to prevent AFW re-initiation

(2) steam dumps

C. (1) lowered to prevent MSL isolation upon initiation of cooldown

(2) intact SG ARVs

D. (1) lowered to prevent MSL isolation upon initiation of cooldown

(2) steam dumps

K/A

0038 Steam Generator Tube Rupture

A2.05 Ability to determine or interpret the following as they apply to a SGTR:

- Causes and consequences of shrink and swell in S/Gs

K/A MATCH ANALYSIS

The question requires the candidate to determine if AFW flow to SGs should be raised or lowered in 19030-C prior to the max rate cooldown and the reason for the flow change. The reason is related to the consequences of shrink and swell associated with the increase steam flow. The candidate is also required to determine the component used to control steam flow - ARVs or steam dumps.

EXPLANATION OF REQUIRED KNOWLEDGE

Following a trip from higher power level, SG levels will shrink to less than the AFW low level actuation setpoint. AFW is then throttled to slowly restore SG levels to approximately 65% during the initial operator actions of 19000-C. With a tube rupture present, AFW to the ruptured SG is isolated as soon as possible using either the approved early operator actions of 10020-C or the isolation steps of 19030-C. Once ruptured SG isolation is complete in 19030-C, a max rate cooldown is performed to establish required subcooling for the subsequent depressurization, which will minimize RCS to SG deltaP. If SG levels are near the AFW low level actuation setpoint when the max rate cooldown commences, the resulting swell and subsequent shrink of SG levels can result in an AFW actuation. The isolated ruptured SG can have AFW flow re-initiated to it, decreasing margin to release to the public. In order to prevent this scenario, 19030-C step 14 directs raising intact SG levels prior to the max rate cooldown to establish levels well above the AFW low level actuation setpoint. Step 14 is a converted note from the original WOG version and the background and bases for this operator knowledge/action is stated in the WOG E-3 background document.

With 1NAA de-energized, only 1 Circ Water Pump is available. In order to utilize steam dumps, which is preferred to minimize release to the public, C-9 must be present and instrument air available. C-9 is determined by one of two Circ Water Pump breakers closed, voltage on the associated pump bus, and condenser vacuum available (14.92" Hg vacuum on one of two instruments on three of three condenser hoods). This interlock is met, so steam dumps will be utilized. Since only the 13.8 KV bus 1NAA was affected and not the RAT, no change to air compressor status is expected - 1NA01 and 1NA05 are energized. Therefore, steam dumps remain available. Per step 17 of 19030-C, ARVs will be used only if steam dumps are unavailable.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per 19030-C step 14 and the associated WOG background, intact SG levels are raised by increasing AFW flow in advance of the max rate cooldown to

prevent re-initiation of AFW flow to the ruptured SG due to the swell and shrink observed during the max rate cooldown.

The second part is incorrect. As discussed in the Explanation of Required Knowledge above, air compressors and C-9 are present and steam dumps are available and will be utilized per step 17 of 19030-C. However, a candidate with insufficient knowledge of plant configuration on a loss of 1NAA could determine that insufficient air compressors or Circ Water pumps are available, resulting in a loss of steam dumps. In this case, ARVs would be utilized.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. As discussed in the Explanation of Required Knowledge above, air compressors and C-9 are present. Steam dumps are available and will be utilized per step 17 of 19030-C.

C. Incorrect. Plausible. The first part is incorrect. Per 19030-C step 14 and the associated WOG background, intact SG levels are raised by increasing AFW flow in advance of the max rate cooldown to prevent re-initiation of AFW flow to the ruptured SG due to the swell and shrink observed during the max rate cooldown. However, a candidate with insufficient knowledge of plant response or of the reason for step 14 could rationalize that AFW flow would need to be lowered to prevent SG level from swelling during the max rate cooldown and causing water flow into the main steam lines, which would require a manual steam line isolation to prevent water hammer.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	038EA2.05
Importance Rating:	2.8 / 2.9

Technical Reference: SOP 13503A-1, Rev 7.2, page 34
EOP 19030-C Rev 39.2, pages 10-15
E-3 WOG Background Rev 2, 4/30/2005
P&ID 1X3D-AA-C02A, Rev 16.0

References provided: None

Learning Objective: LO-LP-37311-07 Using EOP 19030-C as a guide, briefly describe how each step is accomplished.

LO-LP-37311-11 Given a NOTE or CAUTION statement from the EOP, state the bases for that NOTE or CAUTION statement.

LO-PP-18101-24 Discuss the "Shrink" and Swell" in the Steam Generators to include:

a. The conditions that lead to each
b. The possible adverse consequences

LO-PP-28103-05 List all ESF actuation signals with applicable set points, coincidences, permissives, blocks, and discuss the systems response to each ESF actuation signal.

LO-TA-37011 Respond to a Steam Generator Tube Rupture per 19030-C

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 43.5 / 45.13

Comments: Question appears to match the KA.
Need to make sure that EOP Background information is RO knowledge.
The question itself should reference the document being used to support the correct answer:

"In accordance with <19030-C> intact SG AFW flow rates…" and "in accordance with <E-3 WOG Background Document>, the reason…"

There appears to be 2 non-plausible distractors: why would someone choose B (raise AFW flow to intact SGs due to excessive swell) or C (lower AFW flow to intact SGs due to excessive shrink)?
The answer choices do not need to explicitly call out shrink and swell to match the KA. Testing understanding of AFW reinitiation vs. MSL isolation tests the same concept without leading. One possible idea is to ask:
"In accordance with 19030-C step 14, intact SG AFW flow rates are ___(1)____ and then cooldown is initiated per step 17 using ___(2)___."

A. Raised to prevent AFW reinitiation/intact SG ARVs
B. Raised to prevent AFW reinitiation/steam dumps
C. Lowered to prevent MSL isolation upon initiation of the cooldown/intact SG ARVs
D. Lowered to prevent MSL isolation upon initiation of the cooldown/steam dumps

The stem will likely need some more information to bolster the correct answer (i.e., preferred cooldown method given

the conditions of the stem), but the KA is met with the first part of the question and the distractors are all plausible.
-JAT 12/19/2013 (Unsat/Editorial)

New question incorporated above suggestion.
-JAT 2/4/2014

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. Align Steam Dumps for RCS cooldown:	
a. IF Steam Dumps are in T AVG mode, THEN	
1) Match demand on SG Header Pressure Controller PIC-507 and SD demand meter UI-500.	
2) Transfer Steam Dumps to STM PRESS mode using HS-500C	
b. RCS temperature - GREATER THAN 550°F.	b. Momentarily place HS-0500A and HS-0500B in the BYPASS INTERLOCK position.
c. As RCS cooldown is initiated, hold HS-0500A and HS-0500B in the BYPASS INTERLOCK position until RCS temperature is less than 550°F.	
14. Raise intact SG levels prior to maximum rate cooldown.	
15. Check at least one RCP - RUNNING.	15. Perform the following:
	a. Suspend monitoring ruptured loop T-Cold indication on RCS Integrity Status Tree.
	b. Resume monitoring ruptured loop T-Cold indication on RCS Integrity Status Tree if this procedure is exited.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. Initiate RCS cooldown:	
a. Dump steam to Condenser from intact SG(s) at maximum rate using Steam Dumps by slowly raising demand on PIC-507.	a. Dump steam at maximum rate from intact SG ARV(s). IF no intact SG is available, THEN perform the following: Use faulted SG. -OR- Go to 19131-C, ECA-3.1 SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY DESIRED.
***18. Check if RCS cooldown should be stopped:**	
a. Core Exit TCs - LESS THAN REQUIRED TEMPERATURE.	a. WHEN Core Exits are less than required, THEN perform Steps 18.b and 18.c. Go to Step 19.
b. Stop RCS cooldown.	
c. Maintain Core Exit TCs - LESS THAN REQUIRED TEMPERATURE.	

ATTACHMENT C Sheet 2 of 6

PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

CONTROL INTERLOCKS

Control Interlock	Setpoint/Coincidence	Function
C-1 IR Rod Stop	1/2 IR NIS ≥ 20% Rx Power	Auto/manual Rod Stop* May be manually blocked above P-10
C-2 PR Hi ϕ Rod Stop	1/4 PR NIS ≥ 105 % Rx power	Auto/manual rod stop*
C-3 OTΔT Rod Stop / Runback	2/4 ΔT loops w/in 3% of Rx trip setpoint	Auto/manual rod stop* Initiates turbine runback
C-4 OPΔT Rod Stop/Runback	2/4 ΔT loops w/in 3% of Rx trip setpoint	Auto/manual rod stop* Initiates turbine runback
C-5 15% Turbine Power	PT-505 ≤ 15%	Blocks auto rod withdrawal*
C-7 Load Rejection	≥ 10% in 2 minutes load decrease as sensed on PT-506	Arms Steam Dumps
C-9 Condenser Available	1/2 CW pumps running with 2/2 13.8 kV bus UV relays not energized **AND** 1/2 condenser vacuum sensors ≥ 24.92" Hg Vacuum on 3/3 condensers (LPT hoods A, B, & C)	Indicates condenser available for Steam Dumps
C-11 CBD Full Withdrawal	CBD at 220 steps	Blocks auto rod withdrawal *
C-16 Stop Turbine Loading	Auct Lo NR Tavg ≤ 553 °F **OR** Auct Lo Tavg 20 °F below Tref	Stops any increase in turbine load
C-20 AMSAC Enabled	2/2 Turbine Impulse press ≥ 40%	Automatically enables AMSAC

*** AUTO Rod Withdrawal defeated**

STEP: Check Intact SG Levels

PURPOSE: o To control feed flow to the intact steam generators to prevent
excessive RCS cooldown and steam generator overfill.
o To maintain an adequate secondary side heat sink.
o To identify a previously undetected steam generator tube failure
which could potentially result in steam generator overfill

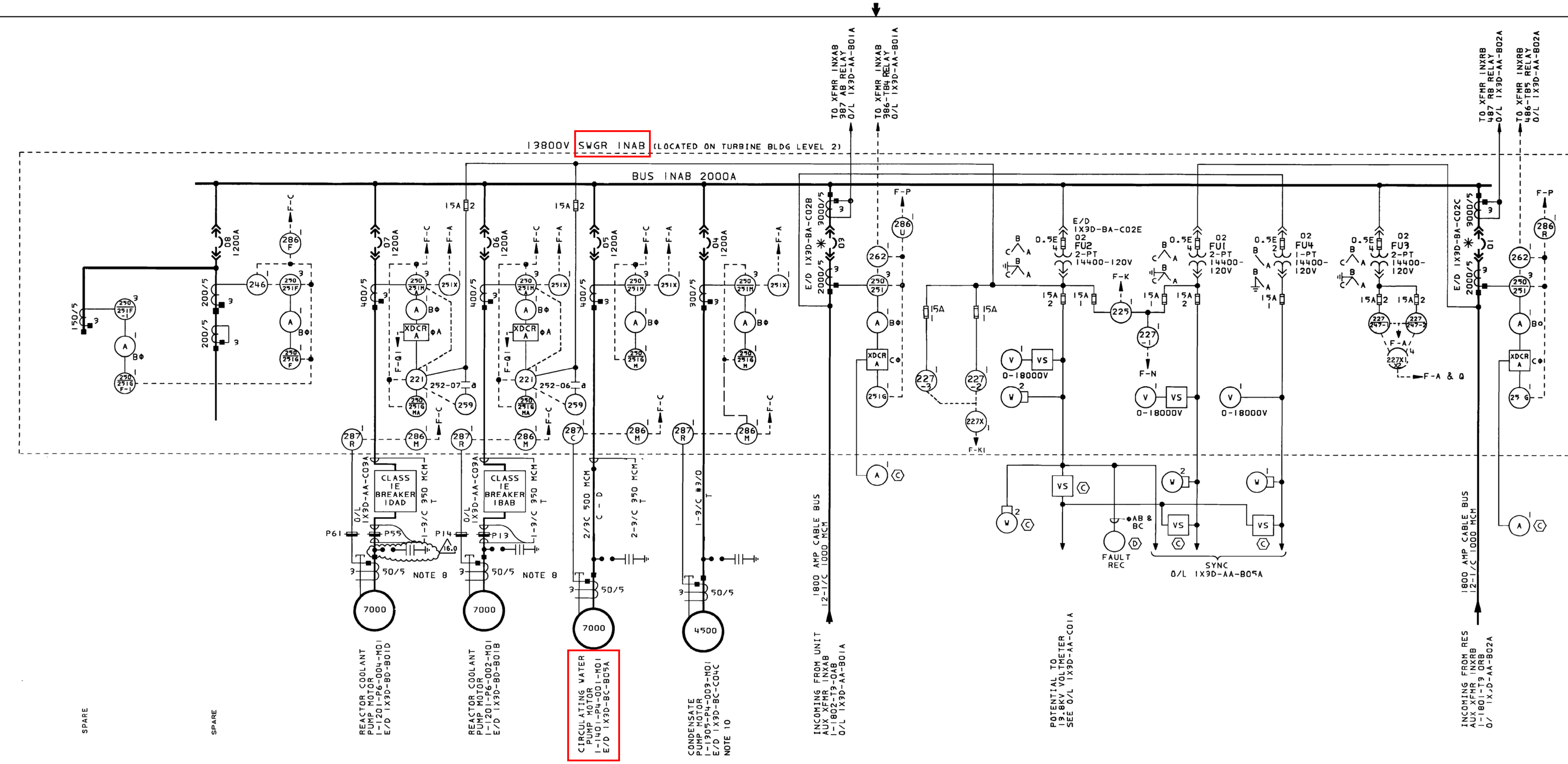
BASIS:

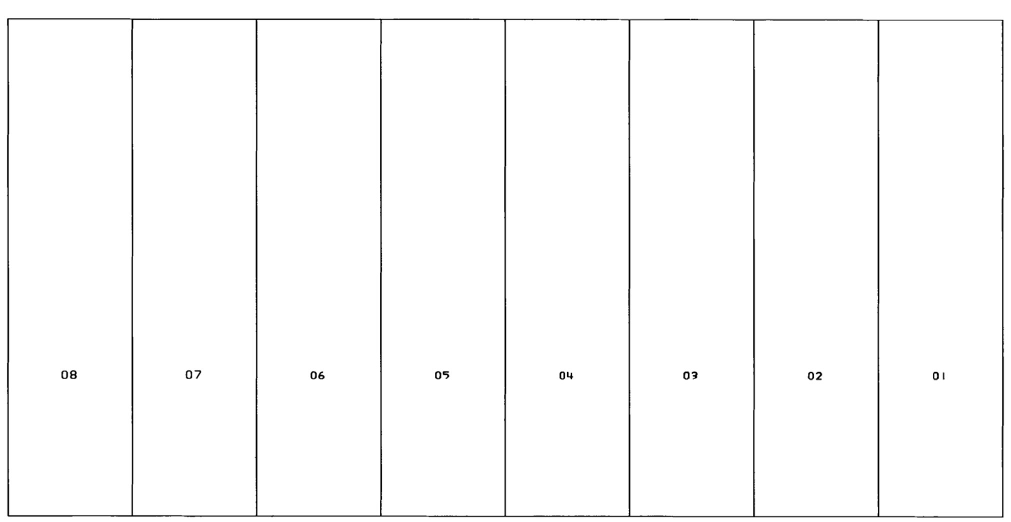
In most cases, feed flow will exceed steam flow from the intact steam generators resulting in an accumulation of water in the steam generators. This excess feed flow will also result in a cooldown of the RCS at a rate dependent upon the feed flow rate and heat generation rate in the primary system. Consequently, feed flow must be adjusted to control steam generator level and reactor coolant temperature. This step also provides for monitoring level in the intact steam generators to detect multiple or subsequent tube failures. In that case, the operator is directed to stop any RCS cooldown in progress and return to Step 1 to isolate the affected steam generator and repeat the recovery actions.

If reactor trip occurs from a high power level, the water level may shrink below the narrow range so that temporarily no reliable indication of steam generator water level is available. During this time, feed flow should be maintained greater than (S.02) to ensure an adequate secondary side heat sink. This minimum feed flow requirement satisfies the feed flow requirement of the Heat Sink Status Tree until level in at least one steam generator is restored in the narrow range. Narrow range level is reestablished in all intact steam generators to maintain symmetric cooling of the RCS. Once intact SG level has been reestablished in the narrow range, the operator is directed to establish a control band between the AFW actuation setpoint and 50%. This control range ensures an adequate inventory will be maintained close to the typical SG level control band and prevent the actuation of the AFW signal. Actuation of the AFW signal could result in potential releases from the ruptured SG through the opened steam

supply valves to the turbine-driven AFW pump if the ruptured SG contained the steam supply tap.

19030-C step 14 does not have a direct step or note listed in the WOG EOP background. It is instead part of the operator knowledge for maintaining SG levels post trip to prevent reinitiation of feed flow once the ruptured SG has been isolated.
Due to operator tendency to forget to prepare for the shrink and swell seen during the max rate cooldown, Vogtle placed a NOTE just prior to the cooldown step to remind operators to raise INTACT SG levels. This prevented SG levels from initially swelling above the low level actuation setpoint and subsequently falling below the setpoint resulting in an re-actuation of AFW and reinitiation of feed flow to the ruptured SG(s). Instead, levels are raised in advance and AFW flow maintained at a relatively high rate during the max rate cooldown.
The NOTE was later converted to step 14 during a HU procedure re-write effort.





FRONT ELEVATION
13800V SWGR 1NAB
SEE NOTE 1

ABBREVIATIONS:

T - TRAY
C - CONDUIT (RIGID GALVANIZED STEEL)
D - UNDERGROUND DUCT
* - 2000 AMP BREAKER

NOTES:

1. RELAY AND INSTRUMENTATION FRONT VIEW LAYOUT OF SWITCHGEAR 1NAB WILL BE SIMILAR TO SWITCHGEAR 1NAA.
2. FOR DEVICE TABLE, GENERAL NOTES, REFERENCE DRAWINGS & LEGEND SEE O/L 1X3D-AA-A00B.
3. DELETED.
4. THE LETTER INSIDE THE HEXAGON INDICATES REMOTE LOCATION OF DEVICE AS FOLLOWS:
   - C ELECTRICAL AUXILIARY BOARD
   - D PLANT FAULT RECORDER CABINET
5. ALL POWER CABLES ENTER SWITCHGEAR FROM THE TOP.
6. FOR LEGEND AND FUNTIONAL TABLE SEE DRAWING 1X3D-AA-C01A.
7. DELETED
8. SURGE PROTECTION ON REACTOR COOLANT PUMPS AS SHOWN ON 1X6AB09-81.
9. DELETED
10. INFRARED WINDOWS (HWK-075-C-L) INSTALLED ON MOTOR JUNCTION BOX.

NUCLEAR SAFETY RELATED

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ONE LINE DIAGRAM
13800V SWITCHGEAR 1NAB
1-1825-S3-0AB

NO.	VERSIONS	DATE	DR	CHK	APPV
16.0	REVISED PER ABN 98-VAN0064, VER 1.0	04-06-10	ELC	TSL	JMR

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-C02A	16.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 039K4.02 001/LOIT AND LOCT/RO/C/A 3.1/3.2/039K4.02/LO-TA-37002/LP-37011///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- Unit 1 reactor trips.
- RTB 'B' fails to open.

Which one of the following completes the following statement?

RCS temperature will be controlled at approximately __(1)__ by the __(2)__.

	__(1)__	__(2)__
A.	557° F	steam dumps
B.	557° F	ARVs
C.✓	559° F	steam dumps
D.	559° F	ARVs

K/A

039 **Main and reheat steam**

K4.02 **Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following:**

- Utilization of T-avg program control when steam dumping through atmospheric relief/dump valves, including T-avg limits.

K/A MATCH ANALYSIS

The question tests the candidate's knowlege of the design and interlocks associated with Steam Dumps, which are part of the Main Steam System, including the response in Tavg mode with a failure of P-4 and the corresponding steady state RCS temperature expected.

EXPLANATION OF REQUIRED KNOWLEDGE

Per 12004-C step 4.1.41.f, ARVs are adjusted to a setpoint of 7.47 during startup. This setpoint corresponds to a RCS temperature of 560F based on saturation. However, each potentiometer has "slop" in the potentiometer and does not control exactly. As such, RCS temperature will normally stabilize between 559F-561F following a reactor trip where the Steam dumps do not arm.

The steam dumps require C-9 and an arming signal from either PT-506 or 'A' Train P-4 to open. Since RTB 'A' opened, the steam dumps received an arming signal. Steam dumps either utilize the Plant Trip or Load Reject controller in Tavg mode. The load reject controller has a 2F dead band and is designed to replace the loss of load while at power by using Tref for a demand signal. When the reactor trips, Tref goes to 557F. The 2F deadband will cause the RCS temperature to stabilize at approximately 559F. The Plant Trip controller has a setpoint of 557F. The Load Reject controller is used when Train 'B' P-4 is not present. Steam dumps transfer to the Plant Trip controller when the RTB 'B' opens.

PT-505 feeds Tref. PT-506 feeds the Steam Dump arming circuit. Both instruments are 1st Stage Turbine Pressure and are side by side on the control board. Both instruments feed into AMSAC.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. RTB 'B' did not open, leaving the steam dumps to control on the Load Reject controller at 559F. However, a candidate that confuses which RTB enables the Plant Trip Controller may believe that steam dumps are controlling on the Plant Trip Controller at 557F.

The second part is correct. Steam dumps are armed by both RTB 'A' and PT-506 lowering on the turbine trip. A candidate who reverses which RTB arms the steam dumps could recognize steam dumps are armed by PT-506 alone and conclude steam dumps are still armed.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Steam dumps were armed by both RTB 'A' and PT-506. ARV's are set to 7.47 and control at 560F and therefore should not open since Steam Dumps should control at a slightly lower temperature. However, candidates often confuse which RTB feeds which function. If the candidate reversed the RTBs and the functions they enable, and the PT-506 arming circuit is overlooked, the candidate could believe that the Steam Dumps did not receive an arming signal and ARVs would open to control temperature.

C. Correct. The first part is correct. RTB 'B' did not open causing the steam dumps to control on the Load Reject controller at 559F.

The second part is correct. See the first part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	039K4.02
Importance Rating:	3.1 / 3.2
Technical Reference:	AOP 18001-C, Rev 35.0, page 43 UOP 12004-C, Rev 107.1, page 35 LOGIC 1X6AA02-00234, Rev 9.0
References provided:	None
Learning Objective:	LO-LP-37011-02 State how the following control systems are employed to automatically stabilize the plant after a reactor trip: a. steam dumps LO-PP-21101-10 Discuss the following concerning the "Atmospheric Relief valves" (ARV): a. Why we have them b. Basic description of how they operate (automatic and manual) LO-TA-37002 Respond to a Reactor Trip Without Safety Injection per 19000-C and 19001-C
Question origin:	MODIFIED - HL17 Question # 041K3.02
Cognitive Level:	C/A
10 CFR Part 55 Content:	41.7
Comments:	

You have completed the test!

1. 041K3.02 001/2/2/STEAM DUMPS-RCS/H 3.8/3.9/NEW/H-17 NRC/RO/SRO/EMT/GCW

Unit 1 initial conditions:

Original Question

- Power is 100%.
- Steam Dumps are in the Tavg mode.

Current conditions:

- Main Turbine automatically trips.
- Reactor Trip Breaker "A" opens.
- Reactor Trip Breaker "B" is CLOSED and cannot be opened.

Based on the current conditions, which one of the following correctly completes the following statement, if no other operator actions are performed?

RCS Tavg will be controlled at __(1)__ due to Steam Dumps controlling on the __(2)__ controller.

	__(1)__	__(2)__
A.	559°F	plant trip
B.✓	559°F	load reject
C.	557°F	plant trip
D.	557°F	load reject

H. FAILURE OF TURBINE IMPULSE PRESSURE INSTRUMENTATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
IMMEDIATE OPERATOR ACTIONS	
H1. Check - NO ROD MOTION.	H1. Place ROD BANK SELECTOR SWITCH in MAN position.
SUBSEQUENT OPERATOR ACTIONS	
H2. Restore TAVG to program band.	
H3. Perform the following:	
a. Verify PIC-507 STEAM DUMP CONTROL set at 1092 psig (approximately 7.28).	Corresponds to a saturation temperature of 557F.
b. Verify PIC-507 in AUTO.	
c. Place HS-500C STEAM DUMP CONTROL MODE SELECT in STM PRESS.	
H4. Check P-7 and P-13 status lights indicate correctly for plant condition within one hour. (TS 3.3.1)	
H5. Initiate the applicable actions of Technical Specification 3.3.1.	
H6. Notify I&C to initiate repairs.	
H7. Initiate the Continuous Actions Page.	

°

INITIALS

NOTE

If the Unit will be held at approximately 30% reactor power, the steam dumps should remain in the steam pressure mode until reactor power is increased. ☐

e. **Transfer** Steam Dumps to Tavg Mode per 13601, “Main Steam System.” ________

f. **Align** each of the Steam Generator Atmospheric Relief Valves for standby by **adjusting** potentiometers to 7.47 and placing controllers in AUTO:

SG1 ARV	PIC-3000A	Pot @ 7.47 and in AUTO	________
SG2 ARV	PIC-3010A	Pot @ 7.47 and in AUTO	________
SG3 ARV	PIC-3020A	Pot @ 7.47 and in AUTO	________
SG4 ARV	PIC-3030A	Pot @ 7.47 and in AUTO	________

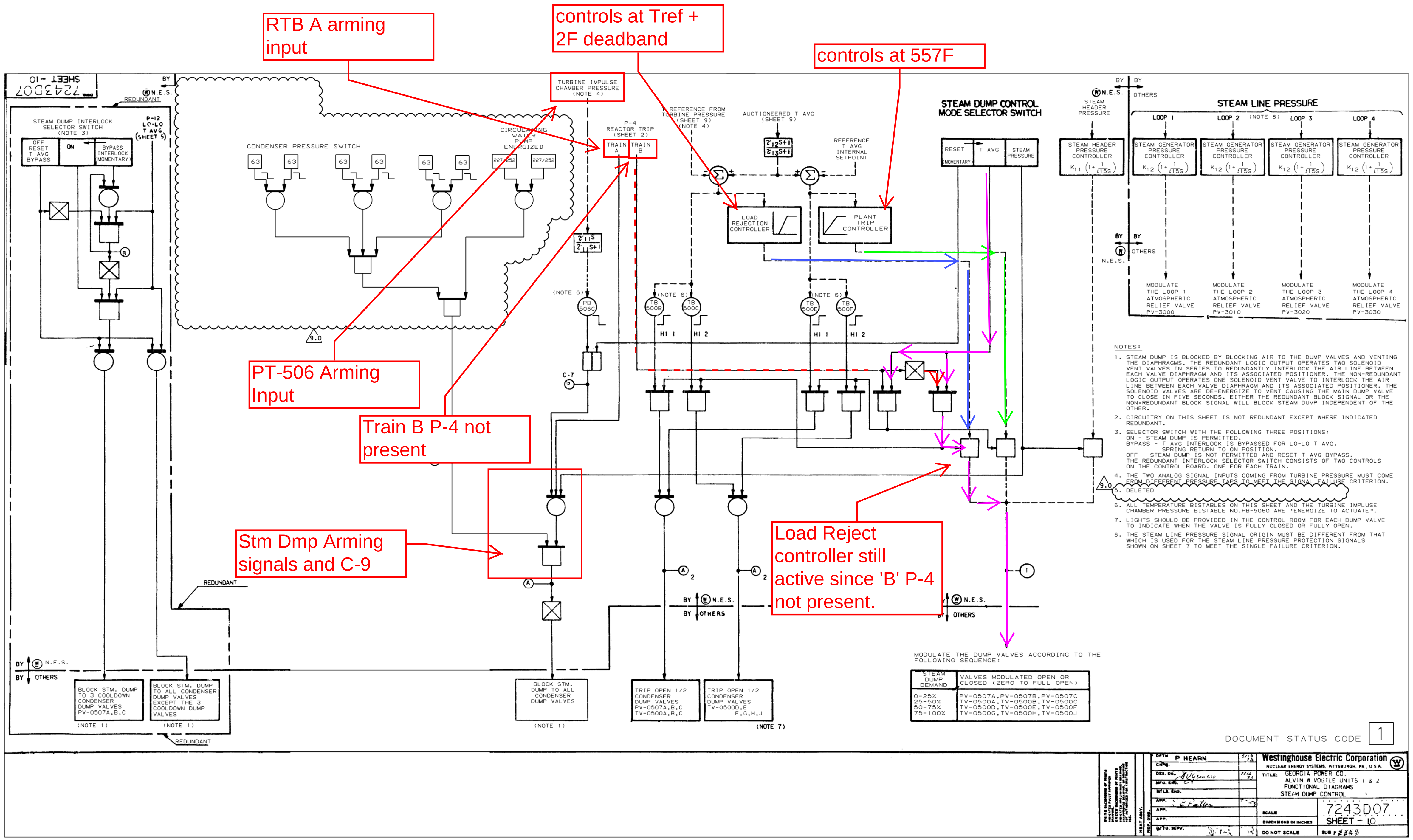
g. **Transfer** sealing steam supply from Auxiliary Steam to Main Steam per 13825, "Turbine Steam Seal System." ________

h. **Transfer** the SJAE Steam Supply from the Auxiliary Steam System to the Main Steam System per 13620, "Condenser Air Ejector System." ________

i. **Perform** calorimetric calibration of Nuclear Instrumentation per 14030, “Nuclear Instrument Calorimetric Calibration."

14030:__________/_______
Date Time
________*

At setpoint of 7.47 corresponds to a pressure of 1120 psig, which corresponds to approximately 559-561F depending on the amount of "slop" in the potentiometer.



THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
TAB/SECT. N/A
PAGE N/A
FIGURE N/A

CAD MICROSTATION DFV/01

1X6AA02-00234.DGN

Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT
UNIT NO. 1

VERSION 9.0	DATE 10-14-05

REVISED BY SNC PER ABN-80649, VER. 1.0

SEE MICROFILM FOR PREVIOUS VER. SIGNATURES.

TITLE: FUNCTIONAL DIAGRAMS
STEAM DUMP CONTROL

BY	CHK'D	APPR.1	APPR.2
DFV	WBJ	ASK	x

VENDOR: WESTINGHOUSE ELEC | P.O.#: –

DRAWING NO. 1X6AA02-00234

1. 040AG2.2.36 001/LOIT AND LOCT/RO/C/A 3.9/4.0/040AG2.2.36/LO-TA-60019///

Initial conditions:

- Unit 1 is at 100% reactor power.
- Train 'A' sequencer simulated software sequence test is in progress per 13540A-1, "Safety Features Sequencer System - Train A."
- Containment Coolers #1, 2, 5, and 6 are running in high speed.

Current conditions:

- A steam line rupture occurs on SGs #1 and 2.
- SG pressures lower from 900 psig to 600 psig in 3 seconds.
- Containment pressure is 14 psig.
- Reactor trip, Safety Injection, and Steam Line Isolation occur.

- NO operator action has been taken.

Which one of the following completes the following statement?

Based on the given conditions, the Steam Line Isolation was initiated by __(1)__,

and

Train 'A' Containment Coolers will be running in __(2)__ speed.

	__(1)__	__(2)__
A.	containment High-2	high
B.	containment High-2	low
C.	SG low pressure	high
D.✓	SG low pressure	low

K/A

040 **Steam line rupture - excessive heat transfer**

G2.2.36 **Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations.**

K/A MATCH ANALYSIS

The question presents a plausible scenario in which conditions are provided in the stem requiring the candidate to determine the system response, taking into account the maintenance activity in progress. The event includes the steam line rupture and has the candidate determine which actuation signal resulted in the equipment lineup stated.

EXPLANATION OF REQUIRED KNOWLEDGE

With 13540A-1 in progress, Sequencer 'A' is in TEST. Some or all actuations are blocked depending on the progression of the test. When the SI signal is received, the Sequencer will automatically exit the TEST mode and run the SI sequence.

Containment Coolers #1, 2, 5, and 6 are running in high speed. These are all Train 'A' components. When the SI sequence runs, the High Speed breaker will trip open and the Low Speed breaker will be closed by the sequencer.

The initiating event for the SI is a steam line rupture on SGs #1 and 2. During this event SG pressures lower from 900 psig (approximately normal pressure for 100% RTP) to 600 psig in 3 seconds. The SI/SLI setpoint is 585 psig on 2/3 SG pressure channels on 1/4 SGs. This circuit is also rate compensated and resulted in a SLI above the setpoint.

Simutaneously, containment pressure rises to 14 psig due to the mass and heat addition. The SLI setpoint is 14.5 psig on 2/3 containment pressure channels. This circuit IS NOT rate compensated and therefore will not result in a SLI.

Note: This scenario was run on the simulator to ensure it was plausbile and the conditions were able to be duplicated exactly as stated.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Containment pressure rises to 14 psig due to the mass and heat addition. The SLI setpoint is 14.5 psig on 2/3 containment pressure channels. This circuit IS NOT rate compensated and therfore will not result in a SLI. However, the candidate may believe this circuit is rate compensated or mistake the HI-1 setpoint of 3.8 psig with the HI-2setpoint and believe a SLI would have occured.

The second part is incorrect. With 13540A-1 in progress,

Sequencer 'A' is in TEST. Some or all actuations are blocked depending on the progression of the test. When the SI signal is recieved, the Sequencer will automatically exit the TEST mode and run the SI sequence. However, SSPS will not automatically exit the TEST mode. A candidate who confuses SSPS and the Sequencer may believe that the Train 'A' Containment Coolers would continue running in High Speed.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. With 13540A-1 in progress, Sequencer 'A' is in TEST. Some or all actuations are blocked depending on the progression of the test. When the SI signal is recieved, the Sequencer will automatically exit the TEST mode and run the SI sequence.

C. Incorrect. Plausible. The first part is correct. The SI/SLI setpoint is 585 psig on 2/3 SG pressure channels on 1/4 SGs. This circuit is also rate compensated and will result in a SLI above the setpoint.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

You have completed the test!

Level:	RO
Tier # / Group #	T1 / G1
K/A#	040AG2.2.36
Importance Rating:	3.1 / 4.2

Technical Reference: SOP 13540A-1, Rev 2.2, page 13
LOGIC 1X6AA02-00231, Rev 8.0
LOGIC 1X6AA02-00232, Rev 17.0

References provided: None

Learning Objective:

LO-PP-21101-06 Describe the operation of the Main Steamline Pressure instruments to include:
a. Where they can be read at (Main Control Room, Shutdown Panels)
b. The number of channels per Main Steamline used by the Reactor Protection System
c. How they function as part of the Reactor Protection System above and below P-11

LO-PP-28201-05 Discuss the sequencer testing operations to include:
a. Simulated Software Testing
b. Selected Output Relay Testing
c. Receipt of a SI or UV signal if testing in progress

LO-LP-37121-05 Describe the plant response to the following conditions:
a. steam line break vs feed line break
b. break at end of life vs break at beginning of life core
c. steam break at full power initially vs zero power
d. feed break inside last check valve vs feed break outside last check valve
e. steam break between SG and first MSIV vs steam break between SG and outside last MSIV

LO-TA-37010 Isolate a Faulted Steam Generator per 19020-C as BOP

LO-TA-60019 Respond to a Loss of Secondary Coolant per 18008-C

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.4 / 41.5 / 41.7

Comments:

You have completed the test!

INITIALS

4.3 SEQUENCER TESTS

NOTES

- If a manual test is in progress, a valid SI or U/V signal will override the test and initiate the accident sequence; therefore, performance of these tests will not render the sequencer inoperable. ☐
- An SI signal will override a U/V signal. ☐
- The following tests may be used to functionally test the Train A Sequencer for return to service following maintenance. Maintenance and/or Engineering should be consulted in determining required testing for returning the sequencer to an operable condition. ☐
- All steps in the following tests are performed at the Safety Features Sequencer Train A Manual Test Panel, unless otherwise specified. ☐

4.3.1 Simulated Software Sequence Test (SI)

NOTES

- The SI SEQUENCE test verifies the proper operation of the SEQUENCER SI timing logic for sequencing SI loads. ☐
- Any alarms received during sequencer testing may be reset by depressing the ALARM ACKNOWLEDGE button on the NORMAL OPERATION OVERVIEW page. ☐

4.3.1.1 **Verify** the Sequencer Train A Timing Logic as follows:

a. On the DIRECTORY page, **verify** the TEST PERMISSIVE yellow indicator is lit. ________

b. **Place** the FUNCTION ENABLE keylock switch to the ACTIVE position (Key 1OP3-41). ________

ALB36-D04 SEQ A IN MANUAL TEST ☐

c. Momentarily **depress** the ENTER TEST MODE button. ________

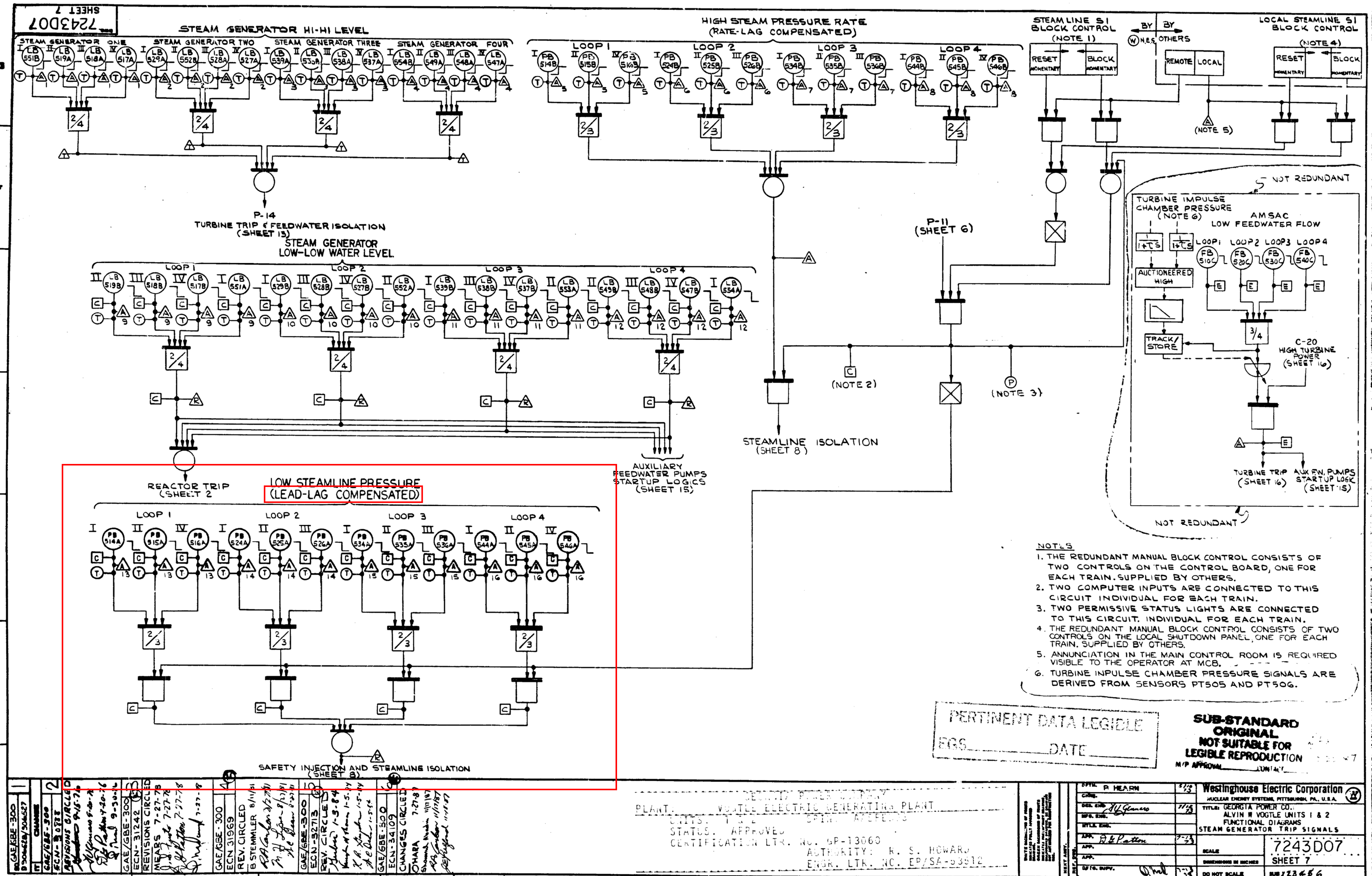
I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL AND REGULAR COURSE OF BUSINESS, ON THE DATE STATED BELOW AND THAT IT IS AN ACCURATE REPRODUCTION OF THE DOCUMENT SUBMITTED TO MICROGRAPHICS.

GPC PLANT VOGTLE CONSTR MICROGRAPHICS

DATE

CAMERA OPERATOR

SECTION SUPERVISOR



BECHTEL NOTE (APPLICABLE TO UNIT 1 ONLY): THIS DRAWING DESCRIBES PHYSICAL WORK TO BE ACCOMPLISHED BY N/A. THIS DRAWING SHOULD BE USED IN CONJUNCTION WITH THE (BWPC NO. N/A) DOCUMENT. THE WORK IS BEING IMPLEMENTED BY PROJECT DOCUMENT.

VOGTLE ELECTRIC GENERATING PLANT — JOB NO. 9510

EQUIPMENT TAG NO. Var

STARTUP DESIGNATION NO. Var

ACTIVITY NO. Var

SYSTEM NO. Var

CATEGORY NO. N/A

RETROFITTING REQUIRED ☐ YES ☒ NO

DISTRIBUTION TO:	FOR: REVIEW	INFO.
MECHANICAL		
HVAC		
EQUIPMENT		
PROCESS		
CONTROL SYSTEMS		✓
ELECTRICAL		✓
WIRING		
CONDUIT		
HAZARDS		
CIVIL/STRUCTURAL		
NUCLEAR		✓
STRESS/PLANT DESIGN		
CODES AND STANDARDS		
ARCHITECTURAL		
STARTUP		✓
CONSTRUCTION		
NOT REQ'D BY ENGRG.		
CLIENT		
EQUIP. QUALIFICATION		
M & QS		
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IDENTIFYING TITLE OF THIS DOCUMENT: Functional Diagrams Steam Generator Trip Signals

Bechtel Log No. 1X6AA02-231-E

IMPORTANT

Permission to proceed does not constitute acceptance or approval of design details, calculations, analyses, test methods or materials developed or selected by the supplier and does not relieve supplier from full compliance with contractual obligations.

DATE RECEIVED 12-9-87

SIGNED

DATE 12-27-87

DOCUMENT STATUS

1 ☒ WORK MAY PROCEED

2 ☐ REVISE AND RESUBMIT WORK MAY PROCEED SUBJECT TO INCORPORATION OF CHANGES INDICATED

3 ☐ REVISE AND RESUBMIT. WORK MAY NOT PROCEED.

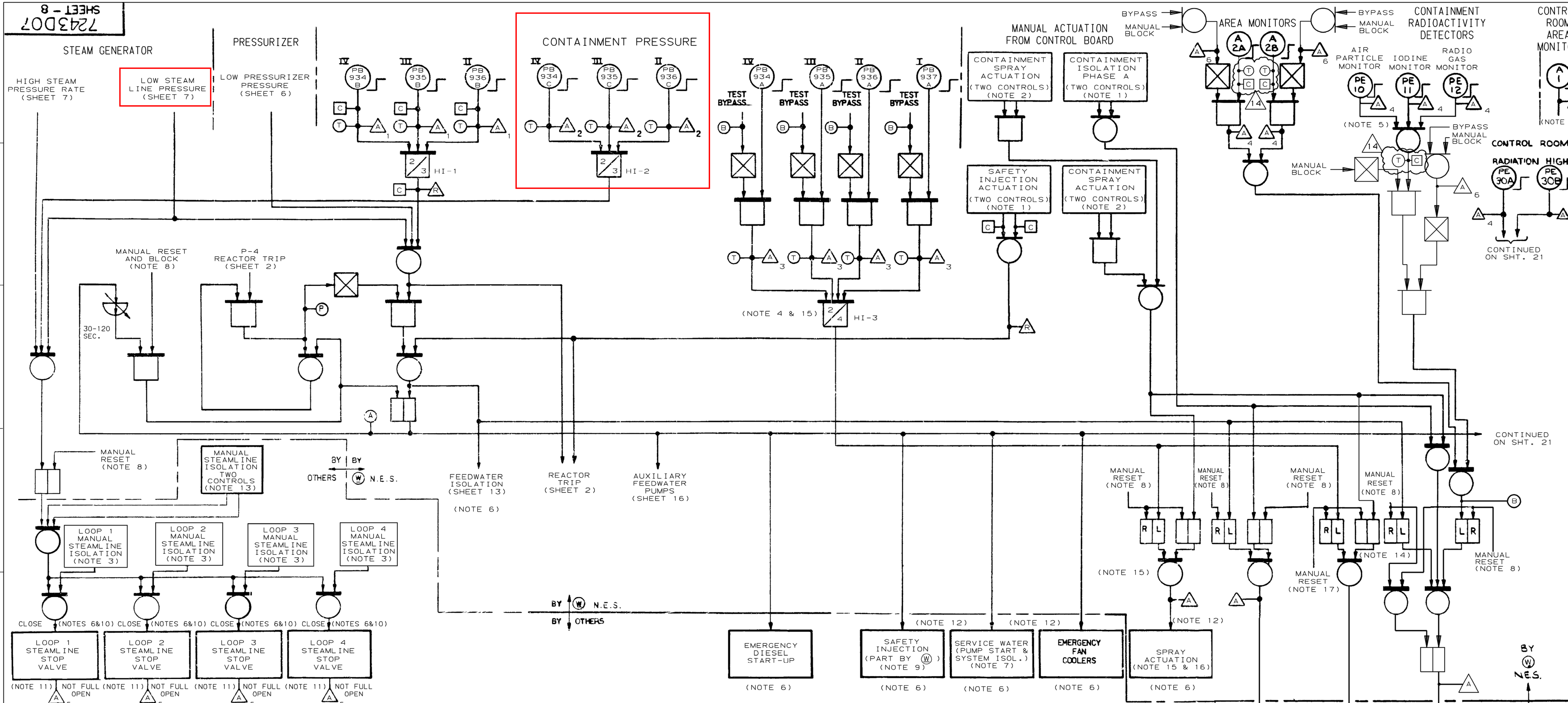
4 ☐ INFORMATION ONLY. DISTRIBUTION REQ'D? ☐ YES

M ☐ RESUBMIT – NOT ACCEPTABLE FOR MICROFILM. WORK MAY PROCEED.

PF-5967 (9510) 12/83

16X

7243D07
SHEET - 8



NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, OPERATING EITHER CONTROL WILL ACTUATE.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS, ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE-TO-ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM, AND IS NOT SAFETY RELATED, HOWEVER, CIRCUITRY IS INPUT TO THE SAFEGUARS SYSTEM THERFORE ISOLATION MUST BE PROVIDED.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
8. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
9. SAFETY INJECTION SEQUENCE REQUIREMENTS (IF SEQUENCING IS NECESSARY) ARE SPECIFIED BY Ⓦ NUCLEAR ENERGY SYSTEMS.
10. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
11. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAM LINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF Ⓦ NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE Ⓦ SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE SEQUENCE TO LOAD THE CONTAINMENT SPRAY SYSTEM ON THE EMERGENCY DIESEL POWER SHOULD BE INDEPENDENT OF THE SEQUENCE TO LOAD THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY THE SEQUENCE INTERLOCK SHOULD BE SUCH THAT SPRAY WILL START WITHIN THE REQUIRED TIME INDEPENDENT OF THE SAFETY INJECTION SIGNAL STATUS.
17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.

14	INCORPORATED PER DCP 97-V1N0067	11-16-00	ELC	TSL	GLB
NO.	REVISIONS	DATE	DR	CHK	APPV

SCS REVISIONS

DOCUMENT STATUS CODE 1

1X6AA02-00232-17

NEXT ASSY. REF. DWG.

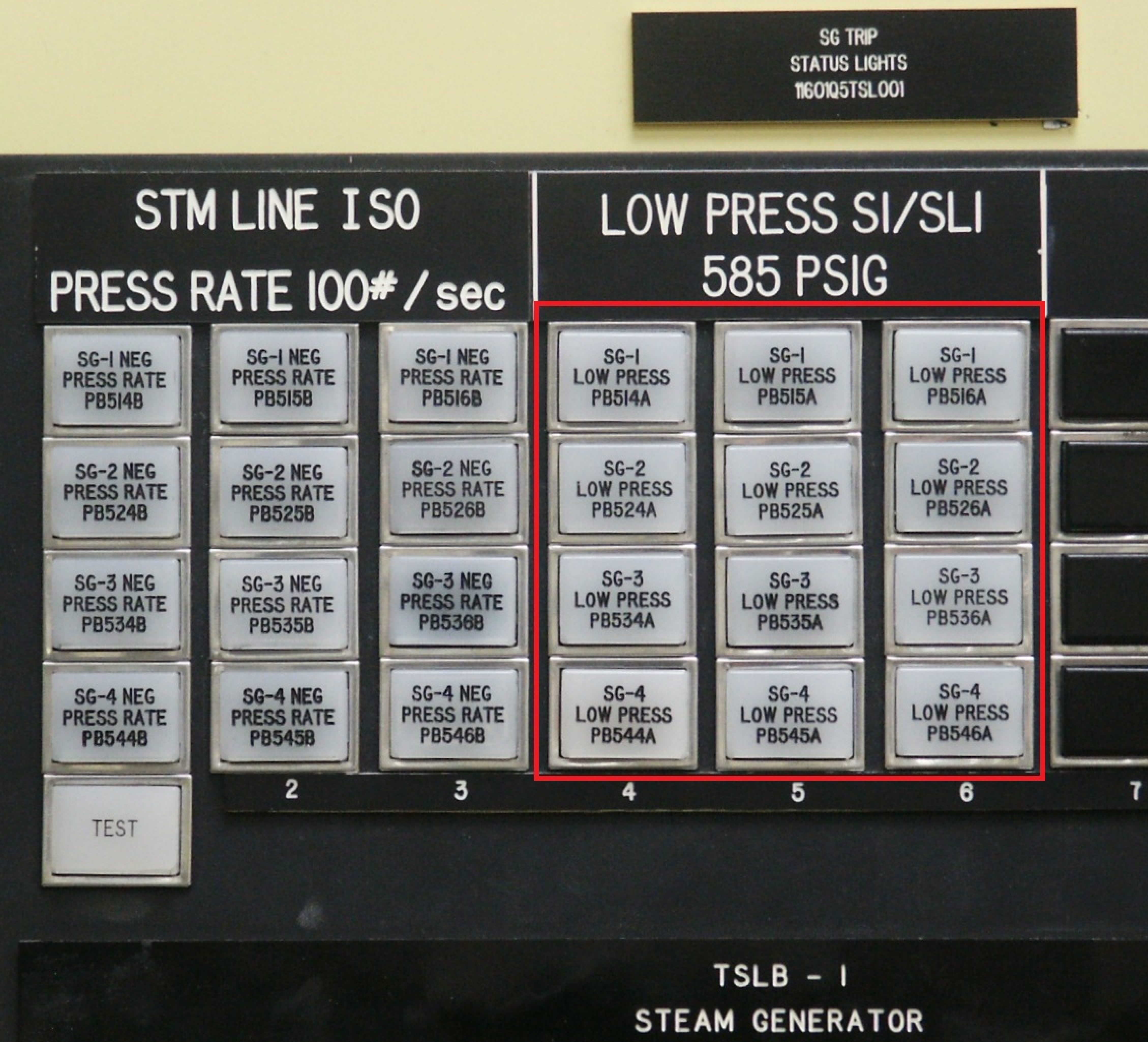
DFTM.	P HEARN	5/10/73
CHKR.		
DES. ENG.	JV Giannaro	7/12/73
MFG. ENG.		
MTLS. ENG.		
APP.	DG Patton	7-12-73
APP.		
APP.		
DFTG. SUPV.		7-13-73

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAM
SAFEGUARD ACTUATION SYSTEM

SCALE
DIMENSIONS IN INCHES
DO NOT SCALE

7243D07
SHEET - 8
SUB 13



SG TRIP
STATUS LIGHTS
1160Q5TSL001
STM LINE ISO
PRESS RATE 100# / sec
LOW PRESS SI/SLI
585 PSIG
SG-1 NEG PRESS RATE PB514B
SG-2 NEG PRESS RATE PB524B
SG-3 NEG PRESS RATE PB534B
SG-4 NEG PRESS RATE PB544B
TEST
SG-1 NEG PRESS RATE PB515B
SG-2 NEG PRESS RATE PB525B
SG-3 NEG PRESS RATE PB535B
SG-4 NEG PRESS RATE PB545B
SG-1 NEG PRESS RATE PB516B
SG-2 NEG PRESS RATE PB526B
SG-3 NEG PRESS RATE PB536B
SG-4 NEG PRESS RATE PB546B
SG-1 LOW PRESS PB514A
SG-2 LOW PRESS PB524A
SG-3 LOW PRESS PB534A
SG-4 LOW PRESS PB544A
SG-1 LOW PRESS PB515A
SG-2 LOW PRESS PB525A
SG-3 LOW PRESS PB535A
SG-4 LOW PRESS PB545A
SG-1 LOW PRESS PB516A
SG-2 LOW PRESS PB526A
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SG-4 LOW PRESS PB546A
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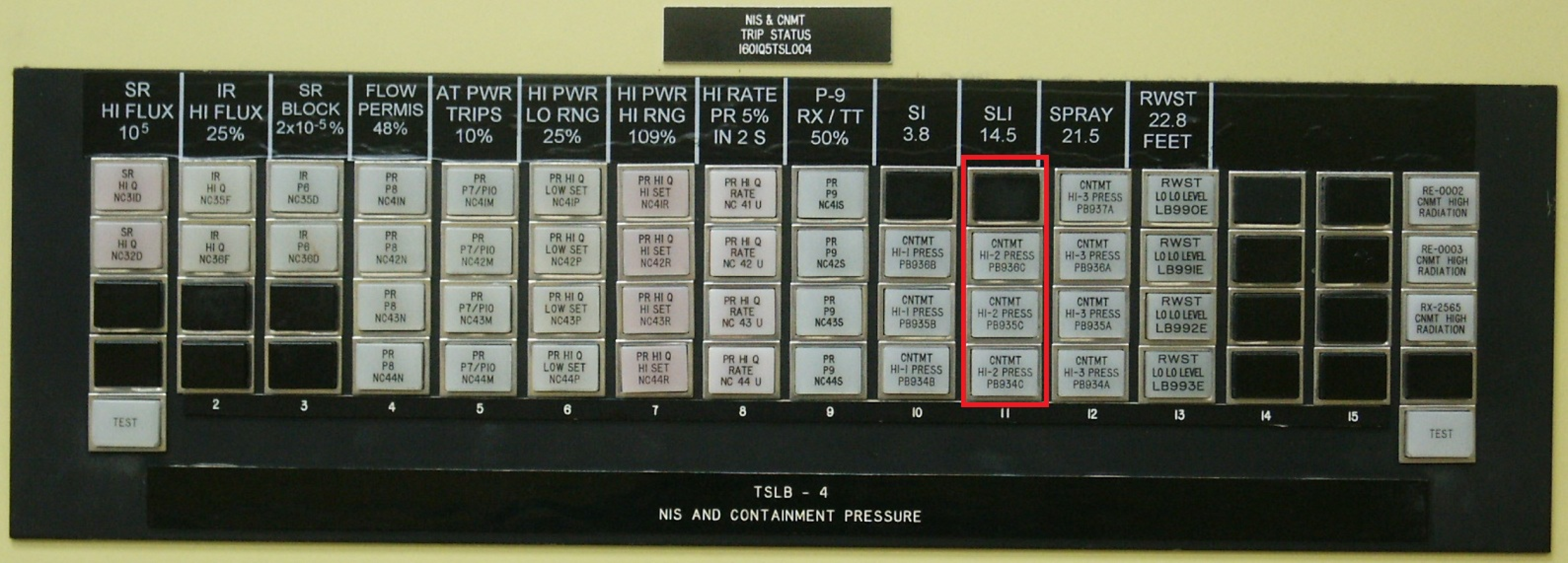
TSLB - 1
STEAM GENERATOR


NIS & CNMT
TRIP STATUS
I60IQ5TSL004
SR HI FLUX 10^5
IR HI FLUX 25%
SR BLOCK $2x10^{-5}$%
FLOW PERMIS 48%
AT PWR TRIPS 10%
HI PWR LO RNG 25%
HI PWR HI RNG 109%
HI RATE PR 5% IN 2 S
P-9 RX / TT 50%
SI 3.8
SLI 14.5
SPRAY 21.5
RWST 22.8 FEET
SR HI Q NC31D
SR HI Q NC32D
TEST
IR HI Q NC35F
IR HI Q NC36F
IR P6 NC35D
IR P6 NC36D
PR P8 NC41N
PR P8 NC42N
PR P8 NC43N
PR P8 NC44N
PR P7/P10 NC41M
PR P7/P10 NC42M
PR P7/P10 NC43M
PR P7/P10 NC44M
PR HI Q LOW SET NC41P
PR HI Q LOW SET NC42P
PR HI Q LOW SET NC43P
PR HI Q LOW SET NC44P
PR HI Q HI SET NC41R
PR HI Q HI SET NC42R
PR HI Q HI SET NC43R
PR HI Q HI SET NC44R
PR HI Q RATE NC 41 U
PR HI Q RATE NC 42 U
PR HI Q RATE NC 43 U
PR HI Q RATE NC 44 U
PR P9 NC41S
PR P9 NC42S
PR P9 NC43S
PR P9 NC44S
CNMT HI-1 PRESS PB936B
CNMT HI-1 PRESS PB935B
CNMT HI-1 PRESS PB934B
CNMT HI-2 PRESS PB936C
CNMT HI-2 PRESS PB935C
CNMT HI-2 PRESS PB934C
CNMT HI-3 PRESS PB937A
CNMT HI-3 PRESS PB936A
CNMT HI-3 PRESS PB935A
CNMT HI-3 PRESS PB934A
RWST LO LO LEVEL LB990E
RWST LO LO LEVEL LB991E
RWST LO LO LEVEL LB992E
RWST LO LO LEVEL LB993E
RE-0002 CNMT HIGH RADIATION
RE-0003 CNMT HIGH RADIATION
RX-2565 CNMT HIGH RADIATION
TEST
2 3 4 5 6 7 8 9 10 11 12 13 14 15
TSLB - 4
NIS AND CONTAINMENT PRESSURE

1. 045K5.17 001/LOIT/RO/C/A 2.5/2.7/045K5.17/LO-TA-61010/LP-61101///

Given the following:

- Unit 1 is at 25% reactor power.
- Power ascension is in progress.

Which one of the following completes the following statement?

As turbine load is increased, main steam header pressure will __(1)__,

and

the method used to maintain Tavg on program during the power increase that results in the MOST negative MTC is __(2)__.

A. (1) decrease

(2) control rod withdrawal with boron concentration held constant

B✔ (1) decrease

(2) boron concentration reduction with control rod position held constant

C. (1) increase

(2) control rod withdrawal with boron concentration held constant

D. (1) increase

(2) boron concentration reduction with control rod position held constant

K/A

045 Main Turbine Generator

K5.17 Knowledge of the operational implications of the following concepts as they apply to the MT/G system:

- Relationship between MTC and boron concentration in RCS as load increases.

K/A MATCH ANALYSIS

The KA is addresses the relationship between changing main turbine load and the expected parameter responses and how operator actions can impact reactor core conditions, specifically how rod and boron changes affect MTC.

EXPLANATION OF REQUIRED KNOWLEDGE

As reactor power increases, RCS Tavg increases due to a ramped program Tavg. The increased RCS Tavg results in a higher SG saturation pressure. However, as the turbine control valves are opened to increase steam flow, SG pressure lowers. The pressure reduction due to increased steam flow is of a larger magnitude than the increase associated with saturation pressure. SG pressure at no load Tavg of 557F is approximiately 1100 psig. SG pressure at full load Tavg of 586.4F is approximately 900 psig.

If the power increase is performed utilizing rods and leaving boron concentration the same, then the boron concentration would be higher at a given power level than if the increase was performed using boron and rod position was unchanged. Per Fundamental Reactor Theory, as boron concentration increases, MTC becomes more positive. Holding rods constant and diluting to power would create the most NEGATIVE MTC scenario.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. As reactor power is increased, steam flow is increased. SG pressure is reduced more by the increase in steam flow than it is increased by raising Tavg. The net affect is a reduction in SG pressure as reactor power increases.

The second part is incorrect. If control rods are withdrawn and boron is held constant, MTC would become more POSITIVE. However, when comparing the change in both rods and boron, it is easy to confuse which one is changing, and think that adjusting boron is actually increasing boron instead of decreasing born concentration.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. If control rods are held constant and boron is changed, MTC would become more NEGATIVE, due to the dilution of the RCS.

C. Incorrect. Plausible. The first part is incorrect. As reactor power is increased, steam flow is increased. SG pressure is reduced more by the increase in steam flow than it is increased by raising Tavg. The net affect is a reduction in SG pressure as reactor power increases. However, if a candidate only considers the change in saturation pressure of the SG and neglects the velocity pressure component, then this answer could be considered correct.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	045K5.17
Importance Rating:	2.5 / 2.7

Technical Reference: None

References provided: None

Learning Objective:

LO-LP-39205-06 State the reason for limitations on MTC.

LO-LP-61101-02 Describe the parameters that change on a load decrease, to include:

a. Turbine power
b. Control valve/CIV positions
c. Steam pressure
d. Steam flow
e. Steam generator level
f. Feed flow
g. Tave
h. Reactor power
i. Rod position (auto mode)

LO-TA-61010 Perform Power Ascent During Low Power Operations using 12004-C

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.1 / 41.5 / 41.14

Comments:

You have completed the test!

1. 054AA1.02 001/LOIT AND LOCT/RO/C/A 4.4/4.4/054AA1.02/LO-PP-20101-04///

At time 1100:

- Unit 1 is at 11% reactor power.
- The Control Room is evacuated due to a fire per 18038-1, "Operation From Remote Shutdown Panels."
- MFP 'A' is tripped prior to exiting the Control Room.

At time 1105:

- Personnel established local control of all components per 18038-1.
- SG NR levels have lowered to 55%.
- Shift Supervisor directs start of all AFW pumps.

Which one of the following completes the following statement?

The MDAFW Pumps __(1)__ require manual start,

and

the TDAFW Pump __(2)__ require manual start.

	__(1)__	__(2)__
A.	will	will
B.	will	will NOT
C.✓	will NOT	will
D.	will NOT	will NOT

K/A

054 **Loss of Main Feedwater**

A1.02 **Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW):**

- Manual startup of electric and steam-driven AFW pumps.

K/A MATCH ANALYSIS

The question tests the candidate's ability to determine if manual starts of the MDAFW and TDAFW pumps is required based on a sequence of events and current plant condtions associated with a Loss of Main Feedwater.

EXPLANATION OF REQUIRED KNOWLEDGE

MDAFW pump automatic start signals are:
- SI
- Trip of both MFPs
- AMSAC
- Lo SG level on 2 of 4 transmitters on 1 of 4 SGs
- LOSP on its associated bus

TDAFW pump automatic start signals are:
- AMSAC
- Lo SG level on 2 of 4 transmitters on 2 of 4 SGs
- LOSP on either bus

Based on plant conditions at 11% RTP, only one MFP is in service. Trip of both MFP actuation circuit is required to be operable prior to raising power above 5% RTP. This is the only MDAFW start signal present. Both MDAFW pumps should be running and would not have to be manually started.

Based on plant conditions, AMSAC is bypassed <40% 1st stage trubine pressure. SG levels are all above 38% NR. No LOSP conditions exist. TDAFW pump should NOT be running and would have to be manually started by the operator.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The MDAFW pumps received an auto start signal from the trip of both MFPs and will NOT require a manual start.

The second part is correct. The TDAFW pump has NOT received an auto start signal and will require a manual start.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. The TDAFW pump has NOT received an auto start signal and will require a manual start.

C.Correct. The first part is correct. The MDAFW pumps received an auto start signal from the trip of both MFPs.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	054AA1.02
Importance Rating:	4.4 / 4.4

Technical Reference: LOGIC 1X6AA02-00231, Rev 8.0
LOGIC 1X6AA02-00231, Rev 11.0
Lesson Plan V-LO-PP-20101, Rev 3.3, slides 45 & 49

References provided: None

Learning Objective: LO-PP-20101-04 List the AFW system automatic start signals and component actuations.
LO-PP-20101-18 Describe the differences between control room and remote shutdown panel operation of the AFW system.
LO-TA-20016 Start the TDAFW Pump from Shutdown Panel "C" using 18038-1/2

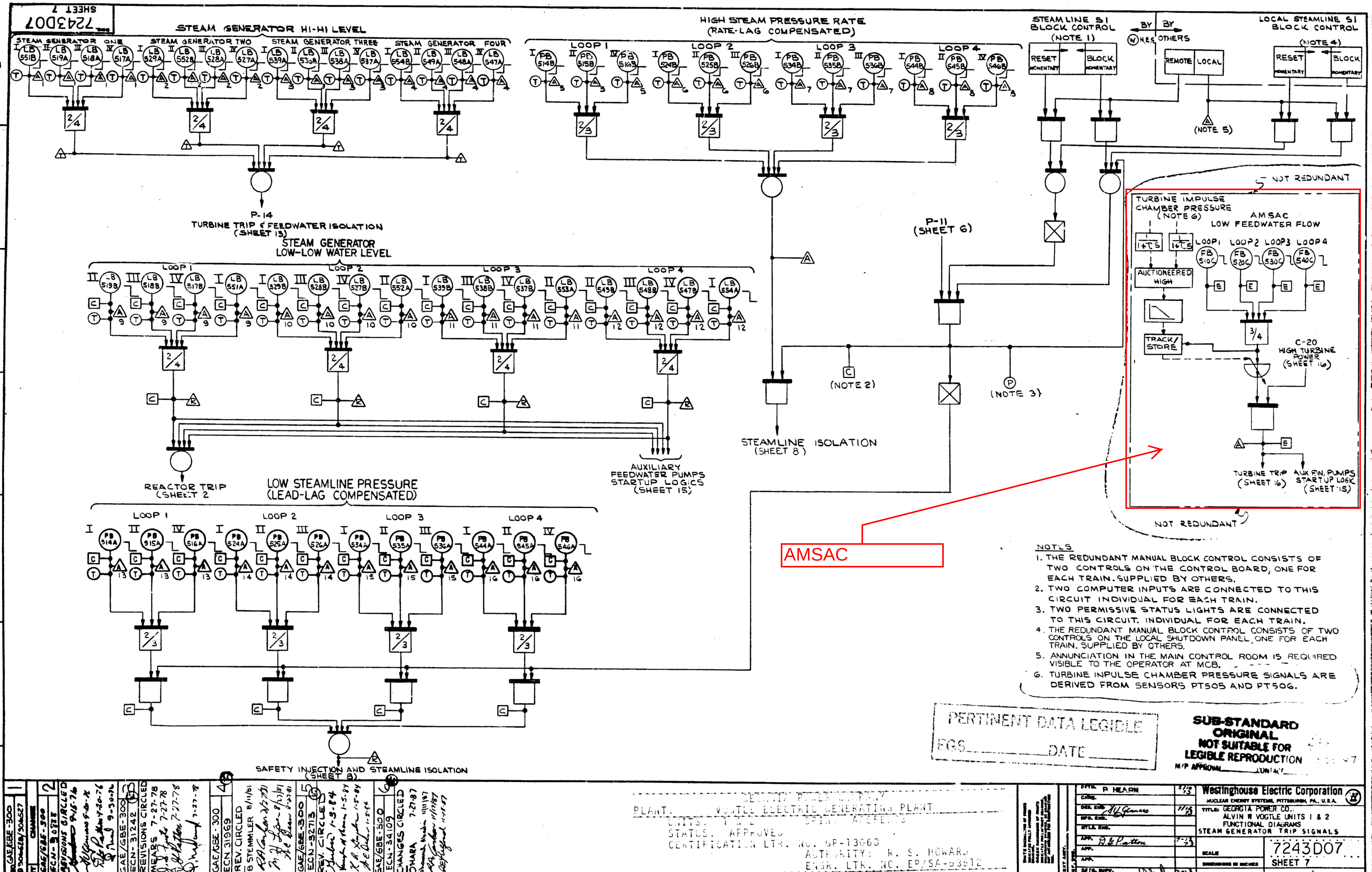
Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.5 / 45.6

Comments:

You have completed the test!



NOTES

1. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN. SUPPLIED BY OTHERS.
2. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT INDIVIDUAL FOR EACH TRAIN.
3. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCUIT. INDIVIDUAL FOR EACH TRAIN.
4. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTROLS ON THE LOCAL SHUTDOWN PANEL, ONE FOR EACH TRAIN. SUPPLIED BY OTHERS.
5. ANNUNCIATION IN THE MAIN CONTROL ROOM IS REQUIRED VISIBLE TO THE OPERATOR AT MCB.
6. TURBINE IMPULSE CHAMBER PRESSURE SIGNALS ARE DERIVED FROM SENSORS PT505 AND PT506.

PERTINENT DATA LEGIBLE
FGS ____ DATE ____

SUB-STANDARD ORIGINAL NOT SUITABLE FOR LEGIBLE REPRODUCTION

PLANT: VOGTLE ELECTRIC GENERATING PLANT
STATUS: APPROVED
CERTIFICATION LTR. NO. GP-13060
AUTHORITY: R. S. HOWARD
ENGR. LTR. NO. EP/SA-53512

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA. U.S.A.
TITLE: GEORGIA POWER CO. ALVIN W VOGTLE UNITS 1 & 2 FUNCTIONAL DIAGRAMS STEAM GENERATOR TRIP SIGNALS
7243D07 SHEET 7
DO NOT SCALE

GPC PLANT VOGTLE CONSTR MICROGRAPHICS
I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL AND REGULAR COURSE OF BUSINESS, ON THE DATE STATED BELOW AND THAT IT IS AN ACCURATE REPRODUCTION OF THE DOCUMENT SUBMITTED TO MICROGRAPHICS.
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SECTION SUPERVISOR
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BECHTEL NOTE (APPLICABLE TO UNIT 1 ONLY): THIS DRAWING DESCRIBES PHYSICAL WORK TO BE ACCOMPLISHED BY N/A. THIS DRAWING (BWPC NO. N/A) SHOULD BE USED IN CONJUNCTION WITH THE REFERENCED DOCUMENT. THE WORK IS BEING IMPLEMENTED BY PROJECT DOCUMENT: DCP-87-V1E0157

VOGTLE ELECTRIC GENERATING PLANT
JOB NO. 9510
EQUIPMENT TAG NO. V.i.r
STARTUP DESIGNATION NO. V.i.r
ACTIVITY NO. V.i.r
SYSTEM NO. V.i.r
CATEGORY NO. N/A
RETROFITTING REQUIRED ☐ YES ☒ NO

DISTRIBUTION TO: FOR: REVIEW INFO.
- MECHANICAL
- HVAC
- EQUIPMENT
- PROCESS
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- HAZARDS
- CIVIL/STRUCTURAL
- NUCLEAR
- STRESS/PLANT DESIGN
- CODES AND STANDARDS
- ARCHITECTURAL
- STARTUP
- CONSTRUCTION
- NOT REQ'D BY ENGRG
- CLIENT
- EQUIP. QUALIFICATION
- M & QS
- WESTINGHOUSE

IDENTIFYING TITLE OF THIS DOCUMENT
Functional Diagrams Steam Generator Trip Signals

Bechtel Log No. 1X6AA02-231-E

IMPORTANT
Permission to proceed does not constitute acceptance or approval of design details, calculations, analyses, test methods or materials developed or selected by the supplier and does not relieve supplier from full compliance with contractual obligations.

DATE RECEIVED 12-9-87
SIGNED
DATE 12-22-87

DOCUMENT STATUS
1☒ WORK MAY PROCEED
2☐ REVISE AND RESUBMIT. WORK MAY PROCEED SUBJECT TO INCORPORATION OF CHANGES INDICATED
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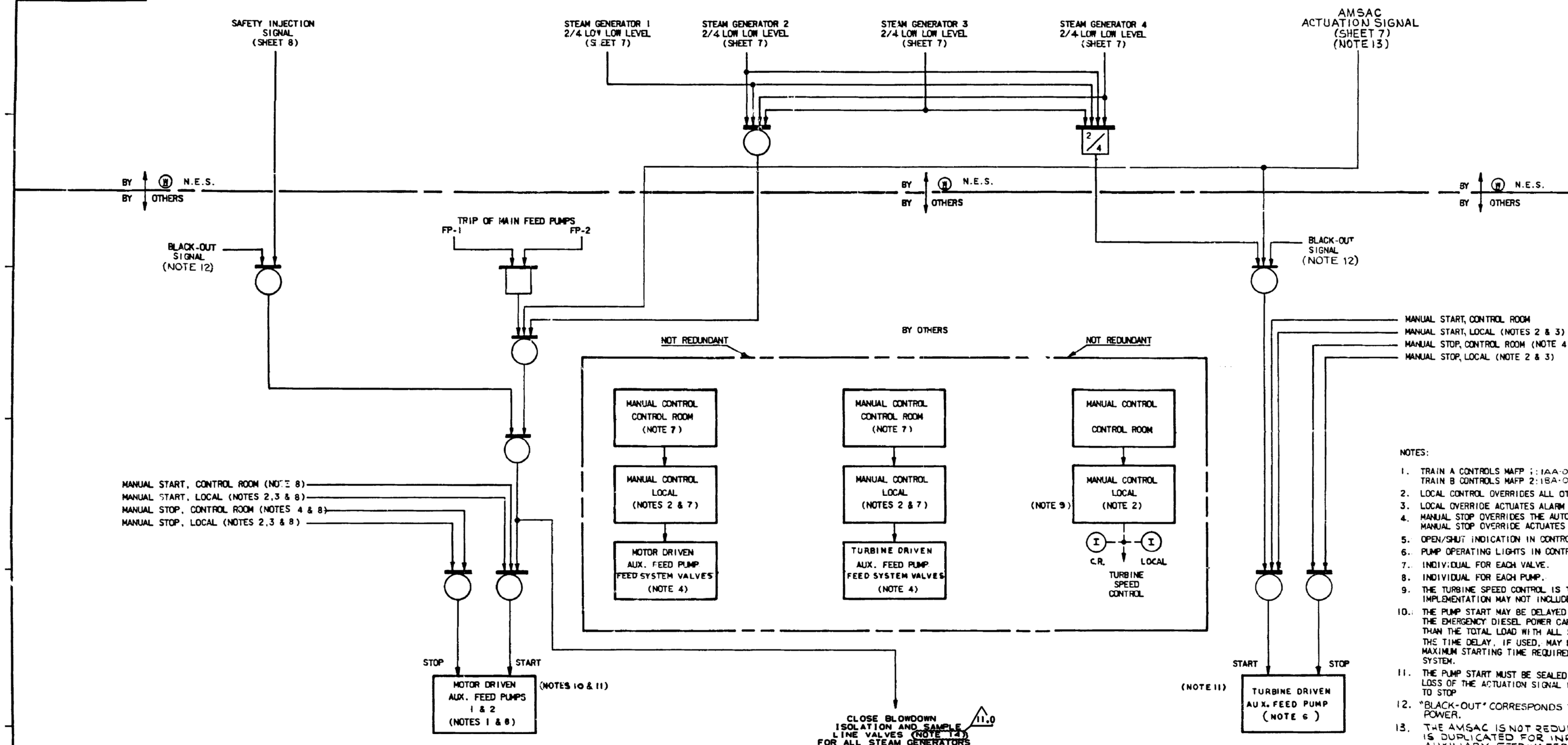
4☐ INFORMATION ONLY. DISTRIBUTION REQ'D? ☐YES
M☐ RESUBMIT – NOT ACCEPTABLE FOR MICROFILM, WORK MAY PROCEED.
PF-5987 (9510) 12/83

16X

7243D07 SHEET - 15



NOTES:

1. TRAIN A CONTROLS MAFP 1: 1AA-02 BREAKER NUMBER 17
   TRAIN B CONTROLS MAFP 2: 1BA-03 BREAKER NUMBER 21
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS.
3. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
4. MANUAL STOP OVERRIDES THE AUTOMATIC START. MANUAL STOP OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
5. OPEN/SHUT INDICATION IN CONTROL ROOM.
6. PUMP OPERATING LIGHTS IN CONTROL ROOM.
7. INDIVIDUAL FOR EACH VALVE.
8. INDIVIDUAL FOR EACH PUMP.
9. THE TURBINE SPEED CONTROL IS TYPICAL. ACTUAL IMPLEMENTATION MAY NOT INCLUDE SPEED CONTROL.
10. THE PUMP START MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY, IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR THIS SYSTEM.
11. THE PUMP START MUST BE SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THE PUMP TO STOP
12. "BLACK-OUT" CORRESPONDS TO A LOSS OF OFFSITE POWER.
13. THE AMSAC IS NOT REDUNDANT. ITS SIGNAL IS DUPLICATED FOR INPUT TO EACH AUXILIARY FEEDWATER ACTUATION CIRCUIT. ISOLATION DEVICES ARE REQUIRED BETWEEN THE NON-1E AMSAC CIRCUITS & THE 1E AUXILIARY FEEDWATER START CIRCUITS.

11.0

14. THE SAMPLE VALVE CIRCUITS ARE TRAIN RELATED. TRAIN "A" CONTROLS HV-9451 AND HV-9454. TRAIN "B" CONTROLS HV-9453 AND HV-9452.

DFTM. P HEARN	5/10/73	Westinghouse Electric Corporation NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.
CHKD.		TITLE: GEORGIA POWER CO. ALVIN W VOGTLE UNITS 1 & 2 FUNCTIONAL DIAGRAMS AUXILIARY FEEDWATER PUMPS STARTUP
DES. ENG. A V Gennaro	7/12/73	
MFG. ENG.		
MTLS. ENG.		
APP. D E Patton	7-13-73	SCALE / 7243D07
APP.		DIMENSIONS IN INCHES / SHEET - 15
APP.		
D' FG. SUPV.	7-13-73	DO NOT SCALE / SUB 1 2 3 4 5 6 7

CAD

MICROSTATION	DFV-01

THIS DWG. IS REFERENCED IN VENDOR MANUAL:

TAB/SECT. N/A
PAGE
FIGURE

Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT
UNIT NO. 1

VERSION 11.0	DATE 4-27-05

REVISED BY SNC PER ABN-80651, VER. 1.0

SEE MICROFILM FOR PREVIOUS VER. SIGNATURES.

TITLE:
FUNCTIONAL DIAGRAMS
AUXILIARY FEEDWATER PUMPS
STARTUP

BY	CHK'D	APPR.1	APPR.2
DFV	PWG	ASK	X

VENDOR: WESTINGHOUSE P.O.#:

DRAWING NO. 1X6AA02-00239

AFW Actuation Signal

- AFW is aligned to be placed in service automatically
  - Motor Driven AFW actuation signal
    - **S**I signal (associated train)
    - **T**rip of BOTH (2/2) MFPTs
    - **A**MSAC signal
    - **L**oss of offsite power (blackout) associated train
    - **L**o-Lo Level in SG - 2/4 detectors on 1 SG

TDAFW Actuation Signal

- Actuating signals
  - **AMSAC signal (From AMSAC Panel)**
  - **Loss of offsite power (either train) (From Sequencer Panel)**
  - **Lo-Lo SG water level - 38% - 2/4 on 2/4 steam generators (From SSPS)**

1. 055EK1.02 001/LOIT AND LOCT/RO/C/A 4.1/4.4/055EK1.02/LO-TA-37018//HL-18 NRC/056AK3.02

Given the following:

- 19100-C, "Loss of All AC Power," is in progress on Unit 1.
- The crew is performing Step 29, "Depressurize intact SGs to 300 psig."
- UO could NOT stop depressurizing SGs at 300 psig.
- All SGs reach 175 psig before the depressurization is stopped.

Which one of the following completes the following statement?

A potential operational implication that could result from the excessive SG depressurization is _____________.

A. an undesired automatic SI signal can occur, which complicates other recovery actions that are in progress

B.✓ nitrogen injection from the accumulators can occur, which disrupts natural circulation flow in the RCS

C. voiding in the reactor vessel can occur, which requires re-pressurization of the RCS before continuing in 19100-C

D. an excessive RCS cool down can occur, which requires transition to 19241-C, "Response to Imminent Pressurized Thermal Shock Condition"

K/A

055 Station Blackout

EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout:

- Natural circulation cooling

K/A MATCH ANALYSIS

The KA addresses the relationship between loss of all AC power and the operational implications of exceeding procedure limits during the natural circulation RCS cooldown.

EXPLANATION OF REQUIRED KNOWLEDGE

Per EOP 19100-C step 29, SGs are depressurized to 300 psig at the maximum rate to mitigate a Loss of All AC. Per the WOG Background ECA-0.0, this action is performed for two reasons. One reason is to minimize the DP across the RCP seals. By lowering RCS pressure, less designed leakage will flow through the seals, delaying seal failure and minimizing inventory loss. Secondly, by rapidly depressurizing the SGs to 300

psig, the SI Accumulators will inject, maximizing RCS inventory. Combining the strategies maximizes mitigation time to allow for recovery of the electrical system.

The max rate depressurization is performed using manual ARV hand pumps. For this reason, stopping the depressurization quickly at 300 psig can be extremely challenging. EOP 19100-C sets a lower limit of 200 psig. This limits ensures the Accumulators will fully inject without injecting nitrogen into the RCS. Nitrogen injection will result in voiding of the SG tubes, disrupting natural circulation and resulting in a loss of Core Cooling.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. 19100-C contains a Continuous Action step to reset a Safety Injection signal that is processed while in this procedure. This step is repeated immediately following the max rate cooldown. The candidate may assume that this is one of the adverse consequences of the maximum rate cooldown that must be avoided. 300 psig is the shutoff head of the RHR pumps and is a pressure commonly associated with ensuring injection flow.

B. Correct. Per EOP 19100-C WOG Background ECA-0.0, if the minimum Steam Generator pressure of 200 psig was exceeded, it could result in loss of natural circulation cooling due to the blocked flowpath as nitrogen voids the SG U-tubes.

C. Incorrect. Plausible. 19100-C has five different criteria for immediately stopping the RCS cooldown, and some of the EOPs dealing with natural circulation has the candidate re-pressurize the RCS to prevent possible interruption of natural circulation flow. (Ref 19002-C step 23 as an example.)

D. Incorrect. Plausible. 19100-C step 30 has criteria to stop RCS cooldown to prevent a PTS challenge; however, this is based on cold leg temperatures instead of Steam Generator pressure. Per WOG Background ECA-0.0 page 141, SG pressure at 300 psig should not challenge this limit. (Note: Tsat for 300 psig is 422F)

Level:	RO
Tier # / Group #	T1 / G1
K/A#	055EK1.02
Importance Rating:	4.1 / 4.4

Technical Reference: EOP 19100-C, Rev 38.1, pages 12, 20 thru 22
EOP 19002-C, Rev 24.0, page 13
WOG EOP Background ECA-0.0, Rev 2, 4/30/2005, pages 24 & 141

References provided: None

Learning Objective:

LO-LP-37031-08	Using EOP 19100-C as a guide, briefly describe how each step is accomplished.
LO-LP-34700-22	Explain the conditions which must exist to establish natural circulation.
LO-LP-34700-23	Describe the means by which the operator can enhance natural circulation.
LO-LP-34700-25	Describe how gas binding affects natural circulation
LO-LP-36101-09	State how the formation of noncondensible gases and/or steam can result in degradation of natural circulation flow .
LO-TA-37018	Respond to a Loss of All AC Power per 19100-C

Question origin: BANK - HL18 NRC Question # 056AK3.02

Cognitive Level: C/A

10 CFR Part 55 Content: 41.2 / 41.8 / 41.10

Comments:

You have completed the test!

1. 056AK3.02 002/1/1/LOSP - EOP ACTIONS/MEM - 4.4/4.7/NEW/HL-18 NRC/RO/SRO/KAJ

Given the following:

Original Question

- A loss of all AC occurred and 19100-C, "Loss of All AC Power," is entered.

- A depressurization of all SGs at the maximum rate is in progress.

Which ONE of the following completes the following statement?

The reason for stopping the SG depressurization at 300 psig is to prevent ________.

A. a steam bubble from forming in the Reactor Vessel Head

B.✔ N_2 injection into the RCS from the ECCS Accumulators

C. challenging the integrity critical safety function

D. a rapid loss of pressurizer level

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. Initiate 13427A/B 4160V AC Bus AA02/BA03 1E Electrical Distribution System to energize at least one AC Emergency Bus using any available power supply: IF offsite power available to either RAT, THEN use normal or emergency incoming feeder breaker. -OR- IF power available to SAT, THEN initiate 13418A(B) Standby Auxiliary Transformer Unit 1(2) Train A(B) Operations. -OR- IF a Diesel Generator extended AOT is in progress AND a Wilson Black Start is necessary, THEN initiate 13419-C Diesel Generator Extended AOT Step 4.3.6.	
	Check for SI and reset is repeated at step 34 following the the max rate cooldown termination steps.
*11. Check for SI:	
a. SI signal - ACTUATED.	a. IF an SI signal exists or is actuated during this procedure, THEN reset SI. IF SI will NOT reset, THEN initiate ATTACHMENT 9. Go to Step 12.
b. Reset SI.	b. IF SI will NOT reset, THEN initiate ATTACHMENT 9.

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. As time permits, perform the following: • Evaluate securing unnecessary battery loads using ATTACHMENT 1. • Initiate 18032-1, 18032-2 Loss of 120V AC Instrument Power and 18034-1, 18034-2 Loss of Class 1E 125V DC Power if the following criteria is met: Any Inverter must be shut down. -OR- Any battery breaker must be opened due to battery overload or low DC Bus voltage.	

NOTES

- The SGs should be depressurized at maximum rate to minimize RCS inventory loss.
- PRZR level may be lost and Reactor Vessel Upper Head voiding may occur due to depressurization of the SGs. Depressurization should not be stopped to prevent these occurrences.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*29. Depressurize intact SGs to 300 psig:	
a. Check SG NR levels - GREATER THAN 10% [32% ADVERSE] IN AT LEAST ONE SG.	a. Perform the following: (1) IF all SG NR levels less than 10% [32% ADVERSE], THEN maintain maximum TDAFW flow. (2) WHEN NR level in at least one SG greater than 10% [32% ADVERSE], THEN go to Step 29.b. Go to Step 33.

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Locally dump steam using SG ARVs at maximum rate: • PV-3000 (South Main Steam Valve Room) • PV-3010 (North Main Steam Valve Room) • PV-3020 (North Main Steam Valve Room) • PV-3030 (South Main Steam Valve Room)	
c. Maintain the following during depressurization: • SG pressures - GREATER THAN 200 PSIG. • SG NR level - GREATER THAN 10% [32% ADVERSE] IN AT LEAST ONE INTACT SG	 • Stop depressurization until level is restored in at least one SG.
*30. Check RCS WR cold leg temperatures - GREATER THAN 280°F [295°F ADVERSE].	*30. Perform the following: a. Control SG ARVs to stop SG depressurization. b. Go to Step 33.
*31. Check SG pressure - LESS THAN 300 PSIG.	*31. WHEN SG pressures lower to less than 300 psig, THEN go to Step 32. Go to Step 33.
*32. Locally control SG ARVs to maintain SG pressures at 300 psig.	

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*33. Check Reactor - SUBCRITICAL: • IR channels - ZERO OR NEGATIVE STARTUP RATE • SR channels - ZERO OR NEGATIVE STARTUP RATE	*33. Control SG ARVs to stop SG depressurization and allow RCS to heat up.
*34. Check SI signal status:	
a. SI - ACTUATED.	a. IF SI actuates during SG depressurization, THEN reset SI. IF SI will NOT reset, THEN initiate ATTACHMENT 9. Go to Step 35.
b. Reset SI.	b. IF SI will NOT reset, THEN initiate ATTACHMENT 9.
35. Check Containment Isolation Phase A using ATTACHMENT 2: Computer Points -OR- Handswitch Indication	35. Actuate Phase A. IF valves do NOT close, THEN locally close at least one valve at each penetration. Locally close any open valve as time permits.
36. Check Containment Ventilation Isolation using ATTACHMENT 3: Computer Points -OR- Handswitch Indication	36. Close dampers and valves. IF dampers and valves can NOT be closed, THEN locally close.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23. Check that steam void in Reactor Vessel does NOT exist: • PRZR level - NO UNEXPECTED LARGE VARIATIONS. • RVLIS Upper Range - GREATER THAN 94%.	23. Repressurize RCS within limits of Technical Specification LCO 3.4.3 (PTLR) to collapse potential voids in system and continue cooldown. IF RCS depressurization must continue, THEN go to 19003-C, ES-0.3 NATURAL CIRCULATION COOL DOWN WITH VOID IN VESSEL (WITH RVLIS).
***24. Check if ECCS should be locked out:**	
a. Check RCS WR pressure - LESS THAN 950 PSIG.	a. WHEN RCS WR pressure is less than 950 psig, THEN go to Step 25. Go to Step 29.
25. Isolate SI Accumulators:	
a. Dispatch an operator to close ACCUM ISO VLV MOV breakers:	

MOV	UNIT 1	UNIT 2
	CB ROOM	CB ROOM
HV-8808A	1ABE-19 (B79)	2ABE-19 (B01)
HV-8808B	1BBC-19 (B61)	2BBC-19 (B18)
HV-8808C	1ABC-19 (B76)	2ABC-19 (B04)
HV-8808D	1BBE-19 (A77)	2BBE-19 (A79)

Step 25 continued on next page

A second limitation to controlled cooldown without ac power is related to the possibility of introducing non-condensible gases into the RCS. Under normal plant shutdown conditions with ac power available, the accumulator injection lines are isolated prior to reducing RCS pressure below 1000 psig. This is done to prevent the accumulator contents from entering the reactor coolant system. In most cases, isolation of the injection lines will not be possible without ac power. Thus, following a total loss of ac event, depressurization to a pressure low enough to allow complete purging of the accumulators must be avoided. In general the operator's only means of doing this will be through controlling the amount of steam being released from the steam generators. Even so, if seal leakage from the RCPs becomes very large, the operator may not be able to control RCS pressure and avoid introducing nitrogen into the system.

The net effect of the limitations on cooldown following a total loss of ac power event is that the operator should be aware of the core criticality concern and how it is affected by fuel burnup. He should understand the desirability of cooling and depressurizing the RCS to reduce the potential for RCP seal failure and to replace lost RCS inventory with accumulator water.

Finally, he should be aware of the limiting low pressure necessary to prevent introduction of non-condensibles from the accumulators. Understanding these considerations, he will be able to depressurize and control secondary pressure to minimize RCS inventory loss while preventing introduction of nitrogen into the RCS and return of the core to a critical condition.

2.4. Transient Analysis Quantitative Results

In order to illustrate quantitatively the system responses to a loss of all ac power as described in the preceding subsection, several computer generated transients representing various scenarios are presented in this subsection. The results presented are for the high pressure (HP) reference plant (i.e., standard 4-loop, 3425 MWt NSSS design); however, standard 2- and 3-loop designs have also been analyzed and determined to exhibit responses similar to those presented herein. Where it is informative, results for the other

During SG depressurization, AFW flow may have to be increased to maintain the required SG narrow range level. Control of AFW flow will have to be performed from the control room or locally depending on plant specific design. Full AFW flow should be established to any SG in which level drops out of the narrow range.

RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function. This check is included in Step 16c since guideline ECA-0.0 has priority over the Function Restoration Guidelines and the operator is instructed to not implement a Function Restoration Guideline even if a Critical Safety Function challenge is detected by the Critical Safety Function Status Trees. Consequently, Step 16c implicitly protects the Integrity Critical Safety Function. The SG depressurization should not result in a challenge to the Integrity Critical Safety Function since the resultant RCS cold leg temperatures should not approach the temperature limit (i.e., T2 temperature) at which a challenge will exist.

Once the target SG pressure is reached, the SG PORVs and AFW flow should be controlled to maintain SG pressure at the target value until ac power is restored.

The target SG pressure for Step 16 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi). To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tanks pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1 V_1^{\gamma} = P_2 V_2^{\gamma}$$

where γ = 1.25 for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta p from P_2 and adding the margin to controllability. The RCS to SG delta p should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the steam generator pressure limit to preclude a bias toward either having more accumulator water injected into the RCS or having less nitrogen injected into the RCS.

1. 056AK1.01 001/LOIT/RO/M/F 3.7/4.2/056AK1.01/LO-TA-37006///

Initial condition:

- Loss of off-site power has occurred.

Current condition:

- 19002-C, "Natural Circulation Cooldown," is in progress.

Which one of the following completes the following statement?

Loop deltaT is expected to __(1)__ as natural circulation flow is FIRST being established,

and

with only 1 available CRDM fan, RCS subcooling is required to be maintained greater than a minimum of __(2)__ during the natural circulation cooldown.

	__(1)__	__(2)__
A.	rise	74°F
B.✓	rise	124°F
C.	lower	74°F
D.	lower	124°F

K/A

056 **Loss of Off-Site Power**

AK1.01 **Knowledge of the operational implications of the following concepts as they apply to Loss of Off-Site Power:**

- Principle of cooling by natural convection.

K/A MATCH ANALYSIS

The KA addresses the relationship between LOSP and natural circulation cooldown. In addition, it refers to the 'principle' of natural circulation, which implies the fundamentals of natural circulation and the things that influence the process. The question has the candidate predict the change in loop delta-T power based on increasing natural circulation flow, and determine, based on the number of CRDM fans available, the amount of RCS subcooling required to prevent Reactor Head voiding and possible interruption of flow. All these are included in the principle of the process and the types of things that change or influence the actions.

EXPLANATION OF REQUIRED KNOWLEDGE

Per WOG Background NAT CIRC page 4, an additional indication of established natural circulation cooling is hot-to-cold leg temperature differences (ie delta T power) approximately equal to full-power forced convection temperature differences.

Per EOP 19002-C step 21, if at least two CRDM fans are running, the RCS subcooling requirement is >74F. If one or no CRDM fans are running, the RCS subcooling requirement is increased to >124F.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. In order to establish the driving head for natural circulation flow, a temperature difference is required between the heat source and heat sink. Per WOG Background NAT CIRC page 4, delta T power should approximately equal full-power forced convection delta T power.

The second part is incorrect. Per EOP 19002-C step 21, if at least two CRDM fans are running, required RCS subcooling is >74F. However, a candidate without specific knowledge of the consequences of inadequate head cooling may assume that the decision between 74F vs 124F is made based on any CRDM fan operating, and not recognize a mimum of two fans is required.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per EOP 19002-C step 21, if at least two CRDM fans are running, required RCS subcooling is >74F.

C. Incorrect. Plausible. The first part is incorrect. In order to establish the driving head for natural circulation flow, a difference in temperature is required between the heat source and heat sink. Per WOG Background NAT CIRC page 4, delta T power should approximately equal full-power forced convection delta T power. However, the candidate may determine that the rising delta T indication implies inadequate heat removal by the secondary and the absence of natural circulation, as seen in a LOHS scenario.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	056AK1.01
Importance Rating:	3.7 / 4.2

Technical Reference: EOP 19002-C, Rev 24.0, page 12
WOG Background NATCIRC, Rev 2, 4/30/2005, page 4

References provided: None

Learning Objective:

LO-LP-37012-05	Using 19002-C, 19003-C, and 19004-C as guides, summarize the actions of these emergency procedures which guide operator response in a natural circulation condition.
LO-LP-37012-15	State the limitations on subcooling and cooldown rate associated with natural circulation cooldown. Include the bases for any variations. (commitment)
LO-TA-37006	Conduct a Natural Circulation Cooldown per 19002-C

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.8 / 41.10 / 45.3

Comments:

You have completed the test!

CONTINUOUS ACTIONS

Step	Actions
2	– Monitor for SI actuation to go to 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
3	– Continue attempts to start an RCP to go to appropriate procedure.
4	– Continue attempts to start CRDM fans as they come available for additional reactor vessel head cooling.
5	– Maintain required boron concentration during cooldown.
10	– Maintain RCS cooldown rate less than 50°F/Hr, SG levels at 65% and RCS temperature and pressure within required limits during cooldown.
18	– Maintain RCS pressure at 1950 psig and PRZR level at 25% after depressurization.
19	– Monitor RCS cooldown.
20	– IFcooldown rate must exceed 50ºF/hour, go to 19003-C, ES-0.3 NATURAL CIRCULATION COOL DOWN WITH VOID IN VESSEL (WITH RVLIS).
21	– Maintain RCS subcooling greater than 74°F (greater than 124°F with less than two CRDM fans) during RCS depressurization.
22	– Maintain RCS cooldown rate less than 50°F/Hr and subcooling greater than 74°F (greater than 124°F with less than two CRDM fans) while continuing RCS cooldown and depressurization.
24	– Monitor RCS pressure less than 950 psig to isolate SI Accumulators.
26	– Monitor RCS WR temperatures less than 350°F to lockout ECCS Pumps.
29	– Maintain letdown flow as necessary.
30	– Maintain seal injection flow to all RCPs 8 to 13 gpm.
32	– Monitor RCS WR cold leg temperatures any less than 220°F to arm COPS.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*21. **Initiate RCS depressurization:**	
a. Verify CRDM Fans - AT LEAST TWO RUNNING.	a. Maintain RCS subcooling - GREATER THAN 124°F. Go to Step 21.c
b. Maintain RCS subcooling - GREATER THAN 74°F.	
c. Verify letdown - IN SERVICE.	c. Depressurize RCS using one PRZR PORV. Go to Step 22.
d. Depressurize RCS using Auxiliary Spray.	
*22. **Continue RCS cooldown and depressurization:**	
a. Maintain cooldown rate in RCS Cold Legs - LESS THAN 50°F/Hr.	
b. Maintain RCS subcooling - GREATER THAN 74°F (124°F with less than two CRDM Fans running).	b. Stop depressurization. Re-establish subcooling.
c. Maintain RCS temperature and pressure - WITHIN LIMITS OF TECHNICAL SPECIFICATION LCO 3.4.3 (PTLR): Use 60°F/HR curve and RCS Cold Leg temperature.	

°

circulation flow are included in the ERGs after SI flow is terminated. If the SI system is in operation, natural circulation flow is not verified since with SI on there are more important steps to be taken and the SI flow may affect the indications used to confirm natural circulation.

If natural circulation flow based on the symptoms listed in Attachment A is not verified, then the ERGs direct the operator to increase steam dump flow to try to establish verifiable natural circulation flow.

The following symptoms are used in Attachment A to verify natural circulation flow:

a) RCS subcooling based on core exit TCs should be greater than instrument inaccuracies.

b) The core exit TCs, RCS hot leg temperatures and SG pressures should be decreasing slowly with time, as core decay heat falls off.

c) With SG pressures held relatively constant, the RCS cold leg temperatures should remain relatively constant at or slightly above the saturation temperature for the SG pressures being maintained.

In addition to the symptoms used in Attachment A, the following symptoms can be used for confirmation of natural circulation flow:

a) The hot-to-cold leg temperature difference should be approximately equal to the full-power forced convection temperature difference.

b) The core exit average temperature (core exit TCs averaged reading) should be higher than the average cold leg temperature. This averaged reading should also decrease as core decay heat falls off, in step with core exit TC, hot leg temperature, and SG pressure readings in all active loops.

To facilitate the verification of transient equilibrium attainment in the natural circulation process, the operators should start to record these parameters at regular intervals beginning as soon as instructed in the ERGs.

1. 057AK3.01 001/LOIT AND LOCT/RO/M/F 4.1/4/4/057AK3.01/LO-TA-37018///055EK1.01

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- An LOSP occurs.
- 19100-C, "Loss of All AC Power," is in progress.
- 1DD1 bus voltage is lowering.

Which one of the following completes the following statement?

Per 19100-C, as bus voltage lowers to less than __(1)__, the operator is required to remove 1DD1 from service using 13431-1, "120 VAC Vital Instrument Distribution System,"

and

the reason for removing 1DD1 from service is to prevent __(2)__.

	__(1)__	__(2)__
A.	100 VDC	battery damage from cell reversal
B.	100 VDC	an explosion due to hydrogen production
C.✓	105 VDC	battery damage from cell reversal
D.	105 VDC	an explosion due to hydrogen production

K/A

057 **Loss of Vital AC Instrument Bus**:

AK3.01 **Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC instrument Bus:**

- Actions contained in EOP for Loss of Vital AC Instrument Bus.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the reason for removing a battery from service when the bus voltage drops below 105VDC during a Loss of All AC.

EXPLANATION OF REQUIRED KNOWLEDGE

Per EOP 19100-C, Step 28, operators monitor 1E bus voltages to ensure they are >105 VDC. If battery voltage drops below 105 VDC, the inverter is removed from service and

the associated battery breaker opened. Per lesson plan V-LO-PP-41201, Slide 18, if battery voltage drops below 90 VDC, cell reversal can occur and permanently damage the cells. (Reference SER 3-99, Vendor Document AX3AD01-00025)

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per EOP 19100-C, Step 28, operators monitor 1E bus voltages to ensure they are >105 VDC. If a battery drops below 105 VDC, the inverter is removed from service and the associated battery breaker opened. However, Lesson Plan V-LO-PP-41201, Slide 7, discusses that the 1E DC system is designed to operate from 140-100 VDC. An exception to this is the inverters, which have a minimum safe voltage of 105 VDC. A candidate without specific knowledge of the procedure step could find both numbers reasonable and, with 100 VDC being 83% of rated voltage, may assume this value is high enough to support the required loads.

The second part is correct. Per lesson plan V-LO-PP-41201, Slide 18, if battery voltage drops below 90 VDC, cell reversal can occur and permanently damage the cells.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per lesson plan V-LO-PP-41201, Slide 18, if battery voltage drops below 90 VDC, cell reversal can occur and permanently damage the cells. However, EOP 19100-C, Step 27, states that all 1E SWGR and inverter room doors must be propped open within 30 minutes to prevent damage due to overheating. Lesson Plan V-LO-PP-41201, Slide 11, discusses how hydrogen gas is produced during the charging process. Battery discharge does not produce as much hydrogen gas, but this concept is commonly reversed by students to justify propping open battery room doors as well as 1E SWGR/Inverter room doors.

C. Correct. The first part is correct. Per EOP 19100-C, Step 28, operators monitor 1E bus voltages to ensure they are >105 VDC. If a DC bus drops below 105 VDC, the inverter is removed from service and the associated battery breaker opened.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	057AK3.01
Importance Rating:	4.1 / 4.4

Technical Reference: EOP 19100-C, Rev 37.5, page 19
Lesson Plan V-LO-PP-41201, Rev 1.1, slides 7, 11, & 18
Vendor Manual AX3AD01-00025, Rev 6.0, page 30
INPO SER 3-99, page 8 & 9

References provided: None

Learning Objective: LO-LP-60330-11 Describe why inverters are shutdown when associated battery bus voltage drops to 105 VDC.
LO-TA-37018 Respond to a Loss of All AC Power per 19100-C

Question origin: BANK

Cognitive Level: M/F

10 CFR Part 55 Content: 41.8 / 41.10

Comments:

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

Equipment failures and loss of control power may occur if doors are not opened within 30 minutes of onset of loss of AC power.

Room B76 on Unit 1 and room B04 on Unit 2 have two doors with installed door stops.

27. Open all doors that have installed door stops in the following affected unit's Control Building electrical equipment rooms:

<u>UNIT 1</u>
B47, B48, B52, B55, B61, B76, B63

<u>UNIT 2</u>
B26, B29, B31, B36, B04, B18, B30

*28. **Check DC Bus loads:**

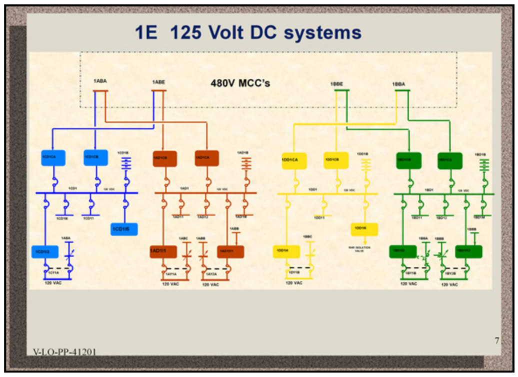
a. Monitor all 1E Battery Bus voltages - GREATER THAN 105V DC.

a. <u>IF</u> any 1E Inverter Battery Bus voltage drops to 105V DC or less, <u>THEN</u> perform the following:

1) Verify associated Inverter(s) shutdown per 13431, 120V AC 1E VITAL INSTRUMENT DISTRIBUTION SYSTEM.

2) Open associated battery breaker after Inverter(s) shutdown.

Step 28 continued on next page



- DC systems utilized as a reliable source of continuous power for control and instrumentation circuits.

- (Review from Electrical Distribution) There are four safety features 125 VDC Trains (designated A, B, C, and D). Each Train has a Battery bank, switchgear, two Battery Chargers, one Inverter (A and B have two), and 125 VDC distribution panels.

-Trains "A", "B", and "C" each have a 125 VDC Motor Control Center for motor operated valves.

-The Batteries provide Emergency power to the DC buses if the Battery Chargers fail. Each Battery is separately housed in a ventilated room (to limit the buildup of Hydrogen) apart from its Chargers and Distribution System and are sized to ensure that all Battery voltages are maintained > 106.2 Volts at the last minute of the **2.75 hour** LOCA/LOSP discharge cycle or at the last minute of the 4-hour Station Black Out duty profile.

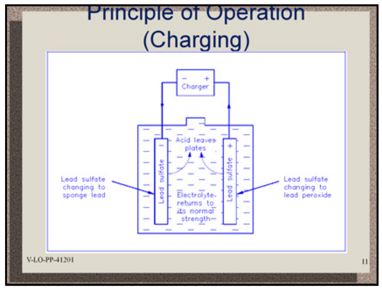
-If a DC over voltage condition is sensed by a Battery Charger, the "Battery Charger Trouble" alarm is annunciated in the Main Control Room.

- All equipment connected to the DC Power System is designed to operate at 140 VDC that exists during the 12-hour period that the Batteries are equalized. All equipment is also designed to operate at 100 VDC except the Inverter Systems that are designed to operate at 105 VDC.

- Control power for many pump breakers will be lost with the DC bus being de-energized and the breakers cannot be remotely operated and protective trips will not open the breaker.

- A Reactor Trip will occur if power is lost to AD1 or BD1 due to the closure of **MSIVs** and **MFIVs**.

- Both chargers normally aligned to the bus and load share. Operating experience showed that with no battery on the bus the chargers will both try to carry the loads and will start developing oscillations. If the electrical bus must be energized by the battery chargers alone (without battery breaker closed in), only **one charger** should be energized to supply the bus.



(Objective – 1)

- As a lead-acid battery is charged, the lead sulfate ($PbSO_4$) is driven off the plates and back into the electrolyte (H_2SO_4). The return of acid to the electrolyte will reduce the sulfate on the plates and increase the specific gravity of the electrolyte. This will continue to happen until all of the lead sulfate is driven from the plates and back into the electrolyte.

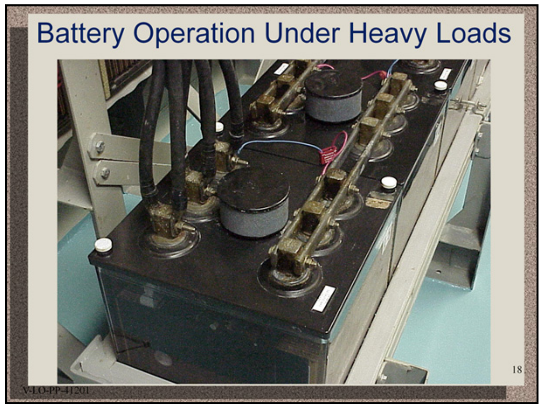
$$2PbSO_4 + 2H_2O \xrightarrow{\text{charge}} PbO_2 + Pb + 2H_2SO_4$$

- A lead-acid battery cannot absorb all the energy from the charging source when the battery is nearing the completion of the charge. This excess energy dissociates water by way of electrolysis into hydrogen and oxygen. Oxygen is produced by the positive plate, and hydrogen is produced by the negative plate. This action (gassing) occurs because the charging current is usually raised in an attempt to drive the remaining lead sulfate off the plates. The excess current also ionizes the water (H_2O) in the electrolyte.

- Excessive gassing is undesirable because it raises the acidity of the electrolyte which could damage the plates, produces an explosive gas, and if the gassing action is violent enough, it can dislodge active material from the grid structure.

- Gassing is first noticed when cell voltage reaches 2.30-2.35 volts per cell and increases as the charge progresses. At full charge, the amount of hydrogen produced is about one cubic foot per cell for each 63 ampere-hours input. If gassing occurs and the gases are allowed to collect, an explosive mixture of hydrogen and oxygen can be readily produced. It is necessary, therefore, to ensure that the area in the vicinity of the battery being charged is well ventilated and that it remains free of any open flames or spark-producing equipment.

- As long as battery voltage is greater than 2.30 volts per cell, gassing will occur and cannot be prevented entirely. To reduce the amount of gassing, charging voltages above 2.30 volts per cell should be minimized (e.g., 13.8 volts for a 12 volt battery).



- 125 VDC batteries response to heavy loads.
- On a loss of AC power (battery chargers), the:
    - 1E batteries are <u>designed</u> to operate at full load for 2¾ hours for a LOCA/LOSP and 4 hours for a SBO (Station Blackout).
    - Non-1E batteries are <u>designed</u> to operate at full load for 2 hours.

- The 1E batteries consist of 59 lead-calcium cells.
- The expected minimum cell voltages for the last minute of the SBO 4-hour profile are as follows:
    - Train A and B - 109.7 battery volts (1.86 volts/cell)
    - Train C - 108.3 battery volts (1.835 volts/cell)
    - Train D - 106.2 battery volts (1.80 volts/cell)
- The longer a battery discharges, the quicker the discharge rate (voltage decay) becomes. The effect is exponential, <u>not</u> linear.
- As loads are stripped battery voltage will take an initial rise and as they are loaded onto the bus the voltage will drop from the present.
- Cell reversal (reverse polarity) will be seen by quick (step change) voltage drops (not caused by load changes). Industry guidance suggests that cell reversal (reverse polarity) may occur when battery voltage drops below 90 VDC.
- Think ahead and be prepared to disconnect the battery from the DC bus by the time battery voltage drops to 100 volts.
- The inverters for AY1A, BY1B, CY1A, and DY1B will not trip until electrical perturbations most likely cause inverter fuse blowing to occur.
- For long-term discharge concerns, a battery can be damaged if voltage is allowed to go down to low.
- It will take about 12 weeks for replacements to be manufactured and delivered.

2.7 Cell Reversal

Cell(s) may be reverse charged during a deep discharge. On recharge these cells may read as high as 4 Volts indicating a high internal resistance. "Reversed cell" temperatures may rise significantly on recharge and immediate action is necessary to avoid permanent damage to the cell(s). Recharge voltage must be decreased to insure that the reversed cell(s) do not exceed 3.0 Volts or 110 degrees Fahrenheit. If the cell(s) do not respond to this special charging it may be advisable to charge the cell individually. A single cell charger is required for this procedure. Contact your C&D agent for additional assistance.

FIGURE 4.2.5 - Cutaway shows clogged separator with the by-products of hydration

2.8 Flame Arrestors, Contamination

If electrolyte levels are permitted to exceed the high level mark at full charge, it is possible for electrolyte to be pumped up into the vent and contaminate the porous stone. In such cases it will be necessary to remove the flame arrester, replacing it with another flame arrester, for cleaning. Adjust the electrolyte level, while the battery is on float charge, to between the high and low level lines.

Clean the contalminated flame arrestor stone in a mild solution of baking soda and water. Provide a final rinse in water and allow the stone to dry. All white deposits should be removed from the stone if the procedure was performed successfully.

2.9 Battery Recycling

When a battery reaches 80% of its rated capacity it is typically considered for replacement. Government regulations require that lead acid batteries at the end of their useful life be beneficially recycled. It is illegal to dispose of industrial lead acid batteries in the trash. C&D provides a recycling program. Contact your local C&D agent for assistance in recycling of your battery.

2.10 Battery records

As noted throughout this manual, battery records are extremely useful for evaluating the installation, operation and maintenance of the battery over its service life. A form is attached (RS-105) to assist you in maintaining a record of service and to be used for warranty records. It is recommended that you make a photocopy of the original so that additional copies may be made for future records.

The initial slow drop in voltage led station personnel to believe that the battery would continue to discharge at a linear rate. Station personnel were unaware that individual cell voltage was a critical parameter for assessing the battery conditions.

Emergency Preparedness

The station declared a Notification of Unusual Event (NOUE) at 9:55 p.m., after losing power to the control room annunciators when the battery discharged. The wording in the emergency action level (EAL) procedure for a loss of off-site power was unclear. With the station blackout relays tripped, the unaffected 480-volt ESF buses could not be realigned to off-site power. The emergency procedures did not adequately address this situation, nor did the EAL table in the emergency preparedness procedures. The EAL procedure was incomplete because it only specified declaring an NOUE if all three sources of off-site power were unavailable. The EAL procedure did not indicate that the basis for the declaration was the inability to provide off-site power to the 480-volt engineered safety features buses. Consequently, operators did not declare an NOUE at 2:50 p.m. when off-site power had been unavailable to the 480 volt ESF buses for greater than 15 minutes.

—Missed Opportunity—
Operators did not recognize that, although off-site power was available, it could not be aligned to the 480-v Class 1E buses, and plant conditions met the technical basis for declaring an Unusual Event.

Some senior reactor operators did not believe that their job responsibilities included evaluating plant conditions with respect to implementation of the emergency plan. Other station personnel, and particularly senior managers, were new to their positions at Indian Point Unit 2 and had not received training on their roles and responsibilities during emergencies.

Consequences

Annunciators Unavailable

At 9:55 p.m., voltage on the No. 24 battery had decreased to approximately 105 VDC and the static inverter supplying power to the No. 24 instrument bus shut down, deenergizing the bus.

Contingency Planning
No formal contingency plans were prepared for the eventual loss of the instrument bus and annunciators

Most of the control room annunciators lost power at this time. The reactor coolant system was still at no-load temperature and pressure, and operations management decided to delay cooling down until the 6A bus and No. 24 instrument bus were restored to service. Station personnel were reluctant to initiate a cooldown with the control room annunciators unavailable because of the difficulties that would be encountered in monitoring plant conditions. Initiating a plant cooldown with the annunciators out of service would have met the conditions for declaring an Alert.

Degradation of the No. 24 Battery

The No. 24 battery was a 125 VDC, 58-cell, lead-calcium battery with an 8-hour rating of 462 amps-hours to 1.81 volts per cell. Technicians began monitoring battery terminal voltage three hours after the battery began discharging. Monitoring did not include determination of individual cell voltages, although industry guidance suggests that cell reversal may occur when battery voltage drops below 90 VDC.

Battery Left On-Line
The battery was allowed to continue discharging after the AC instrument bus lost power

When the No. 24 instrument bus static inverter tripped at 9:55 p.m., the battery had been discharging for 7.5 hours, and battery voltage was less than 100 volts. With the inverter out of service, there was no reason to leave the No. 24 battery connected to its DC bus, but the battery remained in service an additional three hours. Cell reversal occurred at approximately 57 VDC, and battery voltage was 35 VDC when it was disconnected.

Extended Shutdown

Investigation into the incorrectly set overcurrent relays revealed that other nonsafety-related breakers at the station also might have incorrect trip setpoints. As a result, a 45-day unplanned outage was required to inspect and test the plant electrical system.

Effects on Important Systems and Components for Mitigating Core Damage

The Indian Point Unit 2 probabilistic risk assessment (PRA) identifies systems and components based on their relative importance to preventing core damage during postulated accidents. The PRA also identifies risk significant operator actions that may need to be performed to mitigate an accident.

The two most important systems for mitigating events at the station are the auxiliary feedwater system and the pressurizer power-operated relief valves. The two most important operator actions in these events are: starting and aligning the turbine-driven auxiliary feedwater pump (TD-AFWP) following a loss of the motor-driven AFW pumps and initiating reactor coolant system feed-and-bleed cooling using the pressurizer PORVs following a total loss of auxiliary feedwater.

Auxiliary Feedwater System

One motor-driven AFW pump was already unavailable because of the 6A bus outage; however, the one remaining motor-driven AFW pump and the TD-AFWP operated satisfactorily until the No. 24 instrument bus lost power. When the bus deenergized, the flow control valve from the TD-AFWP to the No. 24 steam generator failed full open, as designed.

Additional Risk
Operators challenged the turbine-driven AFW pump by periodically starting and stopping the pump to maintain levels in two of the steam generators.

Operators responded by operating the TD-AFWP in a batch mode, starting and stopping the pump as necessary to maintain adequate level in the No. 23 and 24 steam generators. This posed an additional challenge on the AFW system because the TD-AFWP was more likely to fail when being restarted than it would if operated continuously. The control room crew did not dispatch an operator to manually operate the failed open flow control valve.

Pressurizer Power-Operated Relief Valves

Indian Point 2 operates with both pressurizer power-operated relief valve (PORV) block valves normally closed. Power to one of the block valves was lost when the 6A bus deenergized.

1. 059A4.03 001/LOIT/RO/C/A 2.9/2.9/059A4.03/LO-TA-18018///

Initial conditions:

- Unit 1 is at 30% reactor power.
- MFP 'A' is in service.

Current condition:

- Power is ramping up to 50% at 8% per hour.

Which one of the following completes the following statement?

As power is ramped up, MFP 'A' speed will __(1)__,

and

the BFRVs will continue to throttle open until __(2)__.

A. (1) remain constant

(2) fully open, then remain open

B. (1) remain constant

(2) reaching a prescribed steam flow, then fully close

C. (1) increase

(2) fully open, then remain open

D.✓ (1) increase

(2) reaching a prescribed steam flow, then fully close

K/A

059 **Main Feedwater**

A4.03 **Ability to manually operate and monitor in the control room:**

- Feedwater control during power increase and decrease

K/A MATCH ANALYSIS

The KA addresses the relationship between Reactor Power changes and the feedwater system response, including the expected indications for main feedwater pump speed and feedwater control valve response.

EXPLANATION OF REQUIRED KNOWLEDGE

Per SOP 13506-C Attachment A step 1.5, the Main Feedwater Pump speed will increase as flow demand rises during the power ramp. The feed pump speed demand is calculated as a function of feedwater flow demand. The feed pump speed program accomplishes the following:
(1) Provides adequate pump head to ensure flow to the steam generators is maintained during an expected transient.
(2) Controls feed pump speed and pump head to optimize the position and throttling affect of the feedwater control valves.
(3) Four independent feedwater flow demands (one demand per loop) are calculated by the system. The loop demanding the greatest amount of flow will determine the pump speed demand.

Per SOP 13506-C Attachment A step 1.3, as reactor power rises the BFRVs throttle open to maintain SG levels on program. When the BFRVs reach a demand signal of 50.28 percent the MFRVs receive a signal to throttle open and the MFRVs along with the BFRVs work together to control Narrow Range Level. When Steam Generator Steam Flow reaches 40 % flow rate the BFRVs receive a signal to start closing and will ramp close over an extended period of time (approximately 0.5% per minute of valve demand). The MFRVs will control Narrow Range level from there to 100% power.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per UOP 12004-C, prior to exceeding 5% RTP, the individual slave controller for the inservice MFP and the Master controller for the feed pumps are placed in AUTO. MFP speed will increase as a function of total feedwater demand signal. However, MFPs are in manual while feedwater is transitioned from AFW to the BFRVs. A candidate without sufficient knowledge of MFP operation may assume MFPs are left in manual until a higher power to prevent changes in MFP speed from adversely affecting SG level. This was the case prior to the installation of DFW.

The second part is incorrect. Per SOP 13506-C, BFRVs open until 40% steam flow is reached and the fully close at a rate of 0.5%/min. However, the BFIVs are maitained open after the MFRVs and MFIVs are placed in service. A candidate with insufficient knowledge of the BFRV control circuit may assume that BFRVs are maintained open once MFRVs are placed in service to keep flow through the lines and prevent thermal shock, just like the BFIVs.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. Per SOP 13506-C, BFRVs open until 40% steam flow is reached and then fully close at a rate of 0.5%/min.

C. Incorrect. Plausible. The first part is correct. Per UOP 12004-C, prior to exceeding

5% RTP, the individual slave controller for the inservice MFP and the Master controller for the feed pumps are placed in AUTO. MFP speed will increase as a function of total feedwater demand.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	059A4.03
Importance Rating:	2.9 / 2.9

Technical Reference: SOP 13506-C, Rev 6.0, pages 68 & 70
UOP 12004-C, Rev 108.0, pages 16-20 & 28-29

References provided: None

Learning Objective:

LO-LP-61202-16 Describe the steps to transfer feedwater control from the bypass feedwater regulating valves to the main feedwater regulating valves (Include operator concerns during this transfer).

LO-PP-18101-15 Discuss the operation of the Main Feedwater Pump Turbine SPEED control to include:

a. How the General Electric (G.E.) potentiometer is used to control pump speed
b. How the Westinghouse Controller is used to control pump speed
c. How the Master Controller is used to control pump speed

LO-TA-18017 Shift SG Level control from AFW to the Bypass Feed Regulating Valves using 12004-C

LO-TA-18018 Shift SG Level control from the Bypass Feed Regulating Valves to the Main Feed Regulating Valves

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments:

You have completed the test!

ATTACHMENT A Sheet 2 of 7

The sum of the NR Level Regulator and the Flow Regulator (wide range level feed-forward signal) is used to generate a compensated flow demand for the Valve Lift Calculator.

High Power Mode
The High Power Mode uses Steam Flow/Feedwater Flow Rate-Lag Compensation. This compares the rate at which the flow signals are changing instead of the difference between the signals. The error from the steam flow feed flow rate lag compensation is summed with the NR level error. Since 3 signals are used to create the error this is referred to as 3 element control. The combined error is used for PID control.

Power Mode Transition
The transition between the Low Power Mode and the High Power Mode is based on the measured loop feedwater flow exceeding a predefined threshold.

- During power ascension the transition to high power occurs at 17% measured feed water flow.
- On decreasing power the transition to low power occurs at 14% measured feed water flow.

Both the High Power Mode and the Low Power Mode Track one another which enables a bumpless transfer from one mode to the other.

1.3 Design program for BFRVs and MFRVs during power ascent and descent

As reactor power ascends the BFRVs throttle open to maintain SG levels on program. When the BFRVs reach a demand signal of 50.28 percent the MFRVs receive a signal to throttle open and the MFRVs along with the BFRVs work together to control Narrow Range Level. When Steam Generator Steam Flow reaches 40 % flow rate the BFRVs receive a signal to start closing and will ramp close over an extended period of time (approximately 0.5% per minute of valve demand). The MFRVs will control Narrow Range level from there to 100% power.
During Power Decent the MFRVs will throttle to maintain SG Narrow Range level on program until Steam Generator Steam Flow lowers to < 30% Steam Flow. The BFRVs will throttle open and both the BRFVs along with the MFRVs will control together until the BFRV control signal lowers to 30.28% Demand.
Due to this being a low power level the MFRVs will be manually closed by the operator prior to going below 12 % power to comply with the FSAR 10.4.7.2.3.1.

ATTACHMENT A Sheet 4 of 7

The loss of controllers, force total responsibility of SG Level control for the failed loop(s) onto the operator. Once a controller is restored the M/A Stations return to Manual mode without operator action (either soft or M/A can then be used for control).

As previously stated each M/A Station communicates directly with its RLI I/O module. If a loss of communication between the M/A Station and the RLI I/O occurs the LED indicators on the M/A Station continually flash on and off until communication is restored. When this condition occurs the operator can still use the soft controls at the Operator Work Station to control the affected component.

1.5 Feed pump speed demand

The feed pump speed demand is calculated as a function of feedwater flow demand. The feed pump speed program accomplishes the following:

Provides adequate pump head to ensure flow to the steam generators is maintained during expected transients.

Controls feed pump speed and pump head to optimize the position and throttling affect of the feedwater control valves.

Four independent feedwater flow demands (one demand per loop) are calculated by the system. The loop demanding the greatest amount of flow will determine the pump speed demand.

INITIALS

NOTES

- MFRVs and BFRVs may leakby as the BFIV opens, SG levels should be monitored and AFW flow adjusted to maintain SG levels constant. ☐
- The following step should be completed prior to continuing with Step 4.1.8. ☐

g. **Open** the Bypass Feed Isolation Valve for all SGs one at a time. (1985303297, 1985305760, 1985304988, 1991321521)

- SG 1 HV-15196 ________
- SG 2 HV-15197 ________
- SG 3 HV-15198 ________
- SG 4 HV-15199 ________

INITIALS

CAUTION

During AFW forward flow operations of less than 150 gpm, correct mini-flow valve positions must be maintained. The mini-flow should be checked frequently. ☐

4.1.8 **Transfer** from Auxiliary Feed Water Flow Control Valves to the Steam Generator BFRVs, one Steam Generator at a time, by performing the following: (1985303296 applies to Steps a-h)

a. Using the IPC or OWS, **monitor** flow as control is being transferred on each loop,

		IPC	OWS	
●	SG 1	UF-5404	DFW1FWFLOW	________
●	SG 2	UF-5424	DFW2FWFLOW	________
●	SG 3	UF-5444	DFW3FWFLOW	________
●	SG 4	UF-5464	DFW4FWFLOW	________

NOTE

About 63% or 64% is the preferred SG level when transferring from AFW to the BFRVs. ☐

b. **Stabilize and maintain** the SG NR level(s) between 60% and 70% while transferring to BFRVs. (1985306829) ________

INITIALS

NOTES

- The following step is performed after MFPT control has been transferred from the GE potentiometer to the M/A station per 13615, Condensate and Feedwater Systems. ☐
- By design the 0% to 100% range on the Ovation Speed Demand Signal correlates to a range from 3557 rpm to 6417 rpm. ☐
- When Process Variable is selected, the pump speed or rpm displayed in the digital window will not be accurate until actual pump speed is 3200 rpm or higher. ☐
- The display select button is used to select BIAS, Process Variable or Output in the digital window (BIAS, PV or OUT). ☐
- PV or Process Variable is displayed in RPM and OUT is displayed in percent, or PCT of setpoint demand. ☐
- The display select button may be depressed at any time and as often as necessary to switch parameters in the display window. ☐

c. **Raise** speed of the Feed Pump using the up arrowhead (▲) on MFPT-A(B) controller, SIC-0509B(SIC-509C), to 17.4% output, which is approximately 4050 RPM. ________

NOTES

- Allowing SG level to stabilize slightly below 65% will increase the open output signal to the BFRV causing it to throttle open when placed in AUTO. ☐
- AFW Miniflow valves should be monitored for proper operation when throttling AFW flow. ☐

d. For the SG to be transferred, **adjust** AFW flow until level is maintaining between 63% to 65%.

- SG 1 HV-5139 ________
- SG 2 HV-5132 ________
- SG 3 HV-5134 ________
- SG 4 HV-5137 ________

INITIALS

e. **Place** the BFRV controller in AUTO mode and **check** the BFRV open Output signal increases.

- SG 1 LIC-550 ________
- SG 2 LIC-560 ________
- SG 3 LIC-570 ________
- SG 4 LIC-580 ________

NOTE

Evidence of the BFRV beginning to throttle open should be anticipated between 10 and 20% **output** signal on the M/A Controller. ☐

f. **Close** the Auxiliary Feed Water Supply Valve and **check** the associated BFRV throttles open:

	AFW Supply Valve	BFRV	INITIALS
• SG 1	HV-5139	LV-5243	________
• SG 2	HV-5132	LV-5244	________
• SG 3	HV-5134	LV-5245	________
• SG 4	HV-5137	LV-5242	________

g. **Verify** SG NR level is maintained at program.

- SG 1 ________
- SG 2 ________
- SG 3 ________
- SG 4 ________

INITIALS

h. **Repeat** Steps d. through g. for the remaining Steam Generators:

- SG 1 transferred to Main Feedwater ________
- SG 2 transferred to Main Feedwater ________
- SG 3 transferred to Main Feedwater ________
- SG 4 transferred to Main Feedwater ________

NOTES

- To place the MFP speed controller in auto requires at least one BFRV or MFRV in AUTO. ☐
- ALL steam generator levels should be on program before placing the Main Feedpump in auto to prevent a rapid change in feedpump speed. ☐
- The MFP Output signal should be approximately 17.4% prior to placing its controller in Auto. ☐

i. When ALL steam generator levels are approximately 65%, **place** the **running** MFP controller SIC-509B / SIC-509C in AUTO. ________

NOTES

- Whether in Auto or Manual, the Master Feed Pump Soft Controller will track feed pump slave output which allows a "bumpless transfer." ☐
- Master Feedpump Soft Control is accessed from OWS Graphic 10000. ☐

j. At the Operator Work Station, **place** the MASTER FEEDPMP SPEED Control in AUTO. ________

k. **Check** DFWCS is operating properly and maintaining SG levels at approximately 65%. ________

INITIALS

4.1.22 At greater than 12% Reactor Power, place the MFRVs in service as follows:

NOTES

- MFRVs may leak by as the MFIV opens, SG levels should be monitored and feedwater flow adjusted if required, to maintain SG levels. ☐
- The following annunciator(s) may actuate due to a temperature deviation caused by the cooler water entering the Steam Generators when the MFIVs are opened, ALB13-F03(4,5,6); ☐

  STM GEN 1 (2,3,4) DIGITAL FW CONTROL SYS TROUBLE ☐
- All four MFIVs should be open prior to placing any MFRV in AUTO. ☐
- The following annunciator(s) may actuate while opening MSIV'S, ALB16-A6,ALB16-B6,ALB16-C6,ALB16-D6, MFIV Accum Gas Lo Press. ☐

a. **Verify** feed flow is stable and Steam Generator NR levels are stable between 63% and 67%. ________

b. **Open** the Main Feed Water Isolation Valves for all SGs one at a time: (1985306829, 1984301705)

- SG 1 HV-5227 ________
- SG 2 HV-5228 ________
- SG 3 HV-5229 ________
- SG 4 HV-5230 ________

c. One at a time **place** MFRV Flow Controllers in AUTO and **verify** feed flow and SG levels remain stable:

- SG 1 FIC-510 ________
- SG 2 FIC-520 ________
- SG 3 FIC-530 ________
- SG 4 FIC-540 ________

INITIALS

NOTES

- MFRVs should automatically begin to throttle open when the BFRV setpoint (SP) signal reaches 50.28%. ☐
- The display select button may be depressed at any time and as often as necessary to select SP, PV or OUT in the digital display window. ☐

d. When BFRV's setpoint (SP) signal reaches 50.28%, **check** associated MFRV begins to throttles open:

- SG-1 MFRV FV-510 _________
- SG-2 MFRV FV-520 _________
- SG-3 MFRV FV-530 _________
- SG-4 MFRV FV-540 _________

4.1.23 **Verify** Main Turbine Warming is complete per 13800, "Main Turbine Operation" and **continue** with preparations for Main Turbine Roll. _________

CAUTION

Ensure control of the EX2100 COI touchscreen from the Control Room is established and M1 is selected on the EX2100 COI touchscreen prior to Main Turbine Roll to prevent any delay in generator sync which can lead to high turbine vibrations. ☐

4.1.24 **Verify** Generator/Exciter preparation for startup is complete per 13830, "Main Generator Operation." _________

4.1.25 <u>IF</u> not previously performed, **stop** Sparging Steam to the Condenser per 13615, Condensate And Feed Water Systems." _________

1. 059AK2.02 001/LOIT/RO/M/F 2.7/2.7/059AK2.02/LO-TA-32007///071A1.06 004

Given the following:

- Unit 1 CVCS mixed bed demin resin transfer is in progress.
- A valve failure causes a liquid release on 'C' level of the Auxiliary Building, which results in airborne radioactivity.

Which one of the following completes the following statement?

The airborne radioactivity release will be monitored by __(1)__,

and

the running Auxiliary Building HVAC units __(2)__ automatically trip.

A. (1) ARE-0014, Waste Gas Processing System Effluent Monitor

(2) will NOT

B.✓ (1) 1RE-12442C, Plant Vent Radiogas Particulate (Low Range)

(2) will NOT

C. (1) ARE-0014, Waste Gas Processing System Effluent Monitor

(2) will

D. (1) 1RE-12442C, Plant Vent Radiogas Particulate (Low Range)

(2) will

K/A

059 Accidental Liquid RadWaste Release

AK2.02 Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:

- Radioactive - gas monitors.

K/A MATCH ANALYSIS

The questions tests the candidate's knowledge of Rad Gas monitors and the relationship of these rad monitors to the building HVAC during an accidental gaseous release.

EXPLANATION OF REQUIRED KNOWLEDGE

During normal operation, 2 Aux Building Supply and 3 Aux Building Exhaust Filtration units are in service maintaining negative pressure in the Aux Building envelope. These units are nonsafety-related. During an accidental gaseous release, the effluent is filtered and the exhaust monitored by Plant Vent Rad Monitors RE-12442 and 12444. These monitors are downstream of ARE-12442 and 12444 and have no automatic actuations.

ARE-0014 monitors the release pathway during discharge from the Gaseous Rad Waste system. During a planned gaseous release, ARE-0014 would actuate on receipt of a high rad signal and isolate the release path by closing ARV-0014.

Aux Building Normal Supply and Exhaust units use the Piping Pen Filtration units ductwork during normal operation. On receipt of a CVI signal, the Piping Pen units start and the duct work common to both the Piping Pen and Aux Building Normal systems isolates. Additionally, the Aux Building Normal HVAC units receive a direct trip signal from CVI to ensure these units do not pressurize the Aux Buidling due to the partial ductwork isolation.

CVI is generated by rad monitors RE-002, RE-003, and RE-2565.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. ARE-0014 monitors the release pathway during discharge from the Gaseous Rad Waste system. However, the candidate may determine that, since this is a gas 'release' and A-RE-014 monitors the Auxiliary Building HVAC ducting, it would detect this condition.

The second part is correct. The running Auxiliary Building Exhaust Units will NOT trip as a result of this condition. Aux Building Normal HVAC units receive a direct trip signal from CVI. CVI is generated by rad monitors RE-002, RE-003, and RE-2565.

B. Correct. The first part is correct. RE-12442C is the low range detector that would be used to monitor this type of release.

The second part is correct. See the second part of choice A above.

C. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above..

The second part is incorrect. The running Auxiliary Building Exhaust Units will NOT trip as a result of this condition. Aux Building Normal HVAC units receive a direct trip signal from CVI. CVI is generated by rad monitors RE-002, RE-003, and RE-2565. However, A-RE-0014 does have automatic trip functions which isolate the release flowpath. It is reasonable for a candidate to assume that tripping the exhaust units downstream of this rad monitor might be a suitable secondary measure.

D. Incorrect. Plausible. The first part is correct. See the first part of choice B above.

The second part is incorrect. See the scond part of choice C above.

You have completed the test!

Level:	RO
Tier # / Group #	T1 / G2
K/A#	059AK2.02
Importance Rating:	2.7 / 2.7

Technical Reference: P&ID 1X4DB129, Rev 43.0
P&ID 1X4DB203, Rev 19.0
P&ID 1X4DB208-1, Rev 27.0
P&ID 1X4DB208-2, Rev 17.0
ELEMENTARY 1X3D BG-D02C, Rev 7.0
ELEMENTARY 1X3D BG-D02L, Rev 9.0
SOP 13503A-1, Rev 7.2, page 37

References provided: None

Learning Objective:

LO-PP-23101-02 Describe how the Auxiliary Building Normal HVAC System is operated to perform the following:
a. Minimize the release of radioactive materials to the environment
c. Prevent the buildup of airborne activity level in the Auxiliary Building

LO-PP-23101-03 Describe the response of the Piping Penetration Area Filtration and Exhaust System upon receipt of the CVI signal when initiated by Safety Injection or by Containment High radiation.

LO-PP-23101-04 Describe how the Piping Penetration Area Filtration and Exhaust System is operated to perform the following:
a. Minimize the release of radioactive materials to the environment
b. Prevent the buildup of airborne activity levels in the Piping Penetration Areas and Rooms
d. Maintain a negative pressure in the Piping Penetration Areas

LO-PP-23101-05 Briefly describe how the Piping Penetration Area Filtration and Exhaust System interfaces with the Auxiliary Building Normal Supply and Exhaust System.

LO-PP-29101-21 State any auto actions that occur in the systems listed as a result of the following signals: SI, High Rad, and CVI.
c. Preaccess (normal) Purge
d. Mini Purge

LO-PP-46101-11 State the events that require immediate termination of a gaseous release.

LO-PP-47101-08 Describe the major steps required for

Operations to release a WMT.

LO-PP-29101-12	Describe the importance of RE-12442C and RE-12444C on Containment Mini-Purge.
LO-PP-32101-09	Describe those automatic actions that occur for each of the following non-safety monitors when its high alarm setpoint is exceeded: a. containment vent effluent (RE-2565A, B, C) b. Waste Gas Processing System Effluent (ARE-0014)
LO-PP-46101-03	State the purposes of the following Gaseous Radwaste System components: e. trip valve RV-014 f. rad monitors RE-013 and RE-014
LO-TA-32007	Verify Proper Automatic actions to high radiation alarms

Question origin: BANK - LOIT Question # 071A1.06 004 (HL-15R)

Cognitive Level: M/F

10 CFR Part 55 Content: 41.7 / 41.11 / 41.13

Comments:

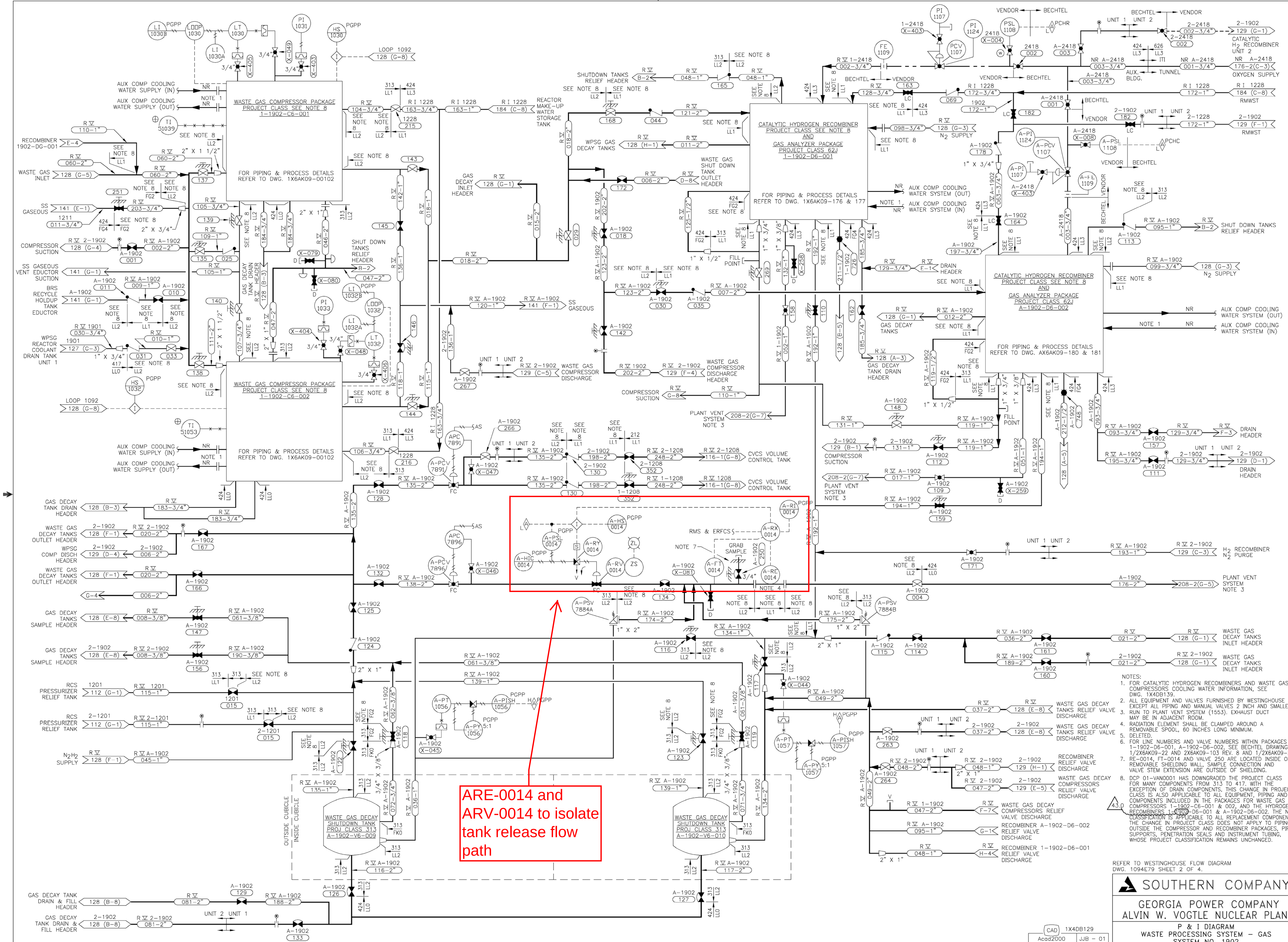
You have completed the test!

ATTACHMENT C Sheet 5 of 6

PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

ESF ACTUATIONS

Actuation	Setpoint		Function/Reset/Bases
AFW Actuation	MDAFWP TDAFWP	Lo Lo Level on 1SG SI (Train related) Trip of Both MFPs LOSP (Train related) AMSAC Lo Lo Lvl on 2 SGs LOSP (either Train) AMSAC	Ensures adequate heat sink. MDAFWP Actuation will also close SGBD isolation valves and SG sample valves. TDAFWP and MDAFWP actuation sends open signal to discharge MOVs. TDAFWP actuation sends open signal to HV-5106. No reset switches but discharge MOVS may be overridden to reduce flow (override indicated by white HS light). Manual actuation consist of manually starting pumps and aligning valves
CIA	SI <u>or</u> 1/2 manual actuation HSs		Isolates non-essential Cnmt penetrations to prevent escape of radioactive material during post-accident conditions. Manual actuation will also result in CVI. May be reset anytime after actuation using Reset HSs (i.e. SI does not have to be reset in order to reset CIA).
CVI	SI Manual CIA Manual CS Hi Rad on RE-002 or 003 (1/2 logic) **<u>OR</u>** Hi Rad on RE-2565 (1/3 channels)		Isolates Cnmt ventilation to prevent escape of radioactive material during post-accident conditions. If initiating signal is SI, then CVI may be reset using HSs on QMCB-A, even if SI has not been reset. If initiating signal was Hi Rad, then HSs on QMCB-C may be used to reset CVI, even if Hi Rad still exist. Either set of Reset HSs may be used if initiating CVI signal has cleared or has been removed/blocked. Hi Rad from RE-002/003 is de-energize to actuate Hi Rad from RE-2565 is an <u>energize</u> to actuate function.
CRI	SI Hi Rad on RE-12116 or RE-12117 Manual on 1/2 HSs on QHVC		Pressurizes CR atmosphere and recirculates air through HEPA filters in order to maintain CR habitable May be reset using Reset HSs on QHVC after initiating signal is cleared. Train B Filter Unit is lead with Train A as lag. Train A will start after time delay if Train B fails to start. Rad monitor input may be manually blocked in QESF. ESF Chilled Water pump start delayed (120 secs on Unit 1, 60 secs on Unit 2) to allow NSCW pressure interlock to be satisfied. ESF Chiller starts after 500 gpm chilled water interlock satisfied.



NOTES:

1. FOR CATALYTIC HYDROGEN RECOMBINERS AND WASTE GAS COMPRESSORS COOLING WATER INFORMATION, SEE DWG. 1X4DB139.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND MANUAL VALVES 2 INCH AND SMALLER.
3. RUN TO PLANT VENT SYSTEM (1553). EXHAUST DUCT MAY BE IN ADJACENT ROOM.
4. RADIATION ELEMENT SHALL BE CLAMPED AROUND A REMOVABLE SPOOL, 60 INCHES LONG MINIMUM.
5. DELETED.
6. FOR LINE NUMBERS AND VALVE NUMBERS WITHIN PACKAGES 1-1902-D6-001, A-1902-D6-002, SEE BECHTEL DRAWING 1/2X6AK09-22 AND 2X6AK09-103 REV. 8 AND 1/2X6AK09-21.
7. RE-0014, FT-0014 AND VALVE 250 ARE LOCATED INSIDE OF REMOVABLE SHIELDING WALL, SAMPLE CONNECTION AND VALVE STEM EXTENSION ARE OUTSIDE OF SHIELDING.
8. DCP 01-VAN0001 HAS DOWNGRADED THE PROJECT CLASS FOR MANY COMPONENTS FROM 313 TO 417. WITH THE EXCEPTION OF DRAIN COMPONENTS, THIS CHANGE IN PROJECT CLASS IS ALSO APPLICABLE TO ALL EQUIPMENT, PIPING AND COMPONENTS INCLUDED IN THE PACKAGES FOR WASTE GAS COMPRESSORS 1-1902-C6-001 & 002, AND THE HYDROGEN RECOMBINERS 1-1902-D6-001 & A-1902-D6-002. THE NEW CLASSIFICATION IS APPLICABLE TO ALL REPLACEMENT COMPONENTS. THE CHANGE IN PROJECT CLASS DOES NOT APPLY TO PIPING OUTSIDE THE COMPRESSOR AND RECOMBINER PACKAGES, PIPE SUPPORTS, PENETRATION SEALS AND INSTRUMENT TUBING, WHOSE PROJECT CLASSIFICATION REMAINS UNCHANGED. 43.0

REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. 1094E79 SHEET 2 OF 4.

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
WASTE PROCESSING SYSTEM – GAS
SYSTEM NO. 1902

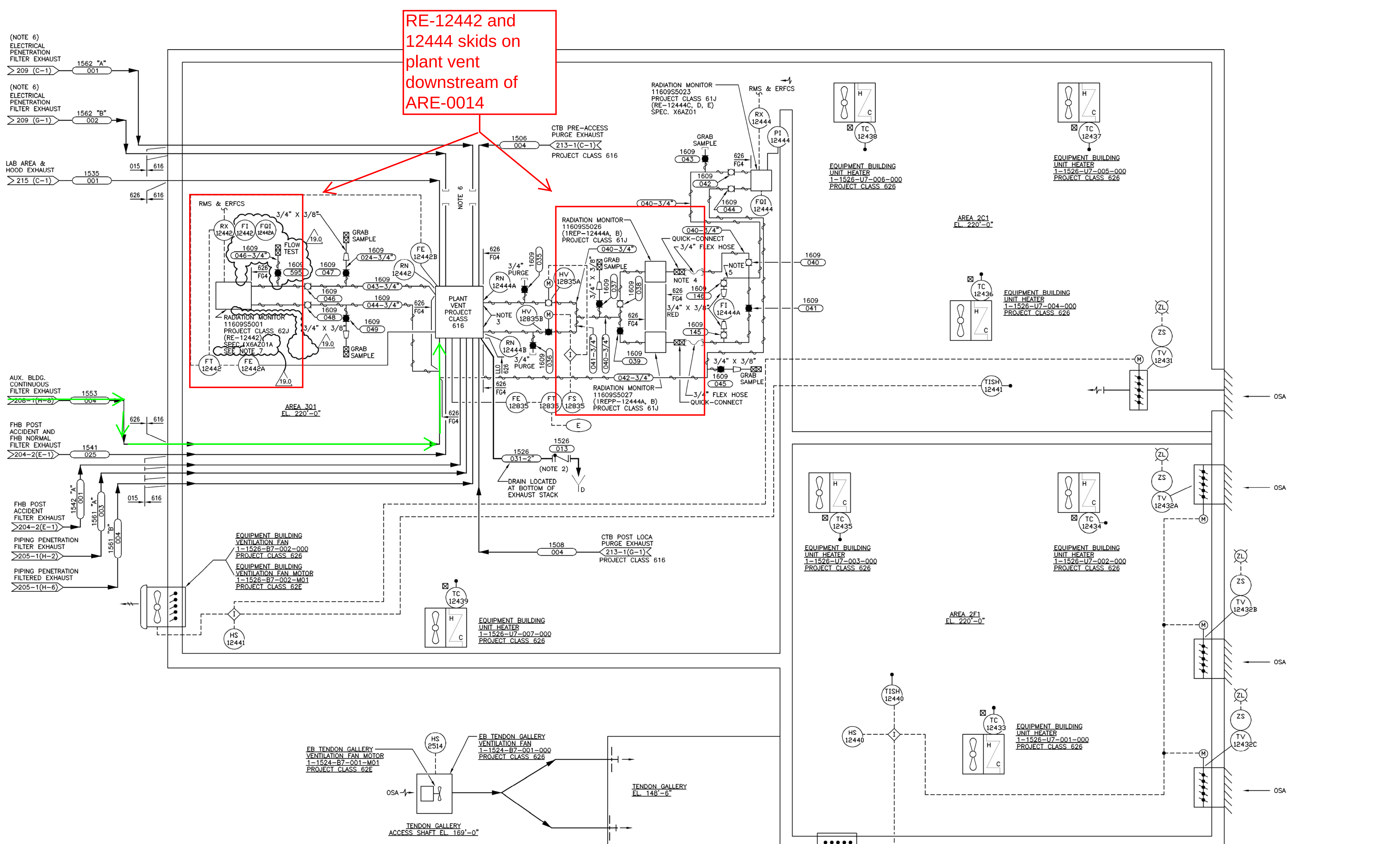
CAD 1X4DB129 Acad2000 JJB – 01

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
43.0	REVISED PER ED SNC364416W001, VER.1.0	2-9-12	JJB	LCF	AAN	X

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X4DB129 | VER. 43.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES: (CON'T)

6. THE CB ELECTRICAL PENETRATION FILTER AND EXHAUST SYSTEM NO. 1562 IS RETIRED-IN-PLACE.
7. FOR INTERNAL PIPING SEE 1X6AZ01A-42001-2

19.0

EQUIPMENT BUILDING VENTILATION FAN 1-1526-B7-001-000 PROJECT CLASS 626

EQUIPMENT BUILDING VENTILATION FAN MOTOR 1-1526-B7-001-M01 PROJECT CLASS 62E

NOTES:

1. ALL DUCTS ARE PROJECT CLASS 626 UNLESS OTHERWISE NOTED.
2. DRAIN CHECK VALVE SHALL BE TECHNOCHECK VALVE OR EQUAL STAINLESS STEEL TYPE. INSTALL HORIZONTAL.
3. ALL INLET LINES FOR AIRBORNE EFFLUENT RADIATION SAMPLE LINES TO EMPLOY 5D RADIUS BENDS.
4. NO 5D BENDS ARE REQUIRED DOWNSTREAM OF 1REP-12444A, B & 1REPP-12444A, B.
5. BYPASS TUBING FOR 1FI12444A TO BE INSULATED (HEAT TRACING NOT REQ'D).

CAD 1X4DB203 ACAD2004 ELC - 02

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
19.0	REVISED PER ABN 1080131801J002, VER. 2.0	09-26-12	ELC	TSL	JMR	X

SOUTHERN COMPANY

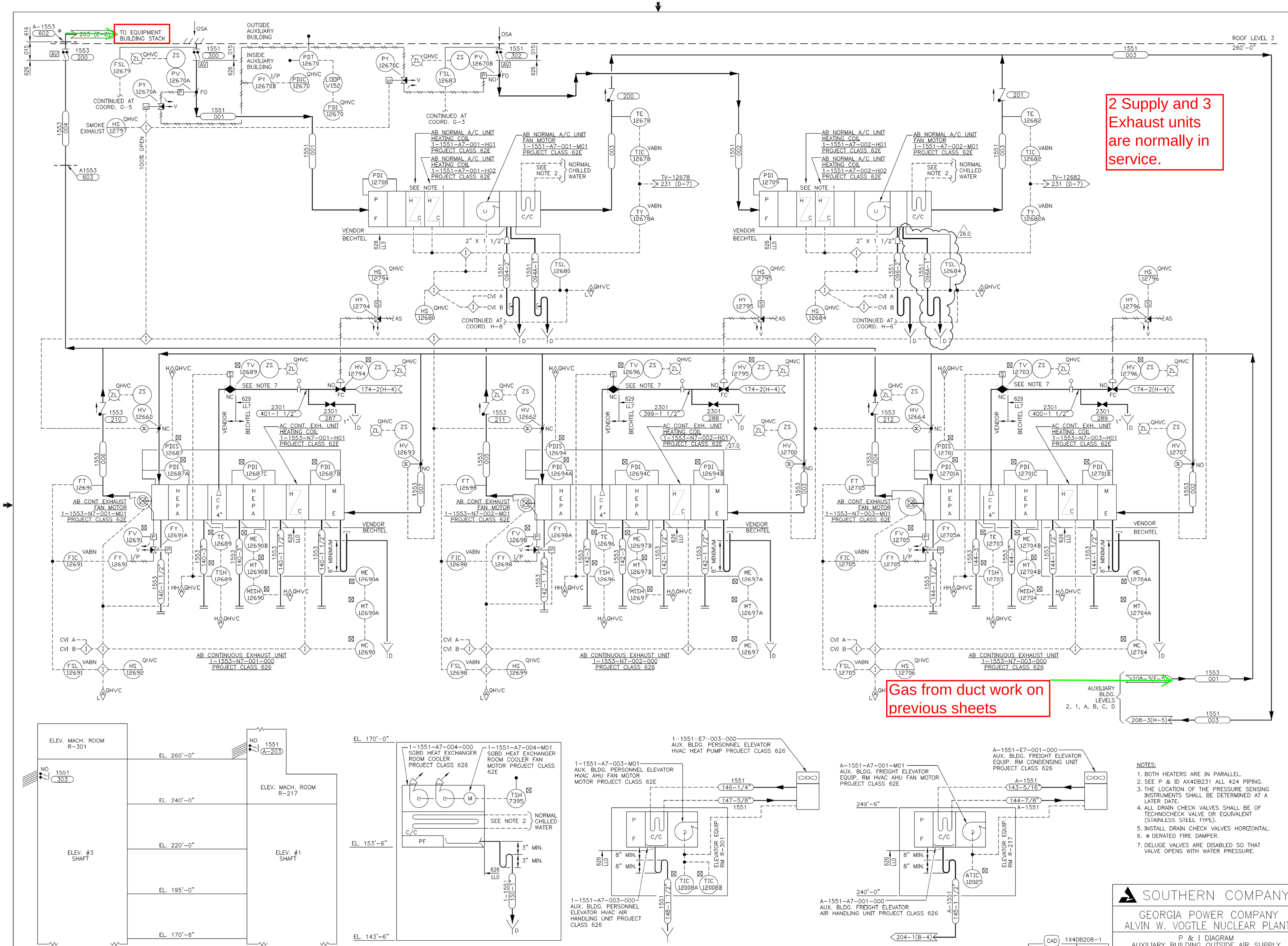
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
EQUIPMENT BUILDING HVAC SYSTEM
SYSTEM NO. 1526 & 1524

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X4DB203 | VER. 19.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. BOTH HEATERS ARE IN PARALLEL.
2. SEE P & ID AX4DB231 ALL 424 PIPING.
3. THE LOCATION OF THE PRESSURE SENSING INSTRUMENTS SHALL BE DETERMINED AT A LATER DATE.
4. ALL DRAIN CHECK VALVES SHALL BE OF TECHNOCHECK VALVE OR EQUIVALENT (STAINLESS STEEL TYPE).
5. INSTALL DRAIN CHECK VALVES HORIZONTAL.
6. * DERATED FIRE DAMPER.
7. DELUGE VALVES ARE DISABLED SO THAT VALVE OPENS WITH WATER PRESSURE.

NO.		DATE	DR	CHK	APPV	DTL
27.0	REVISED PER ABN -V02809, VER. 1.0	12/07/11	PVK	DRW	AAN	X

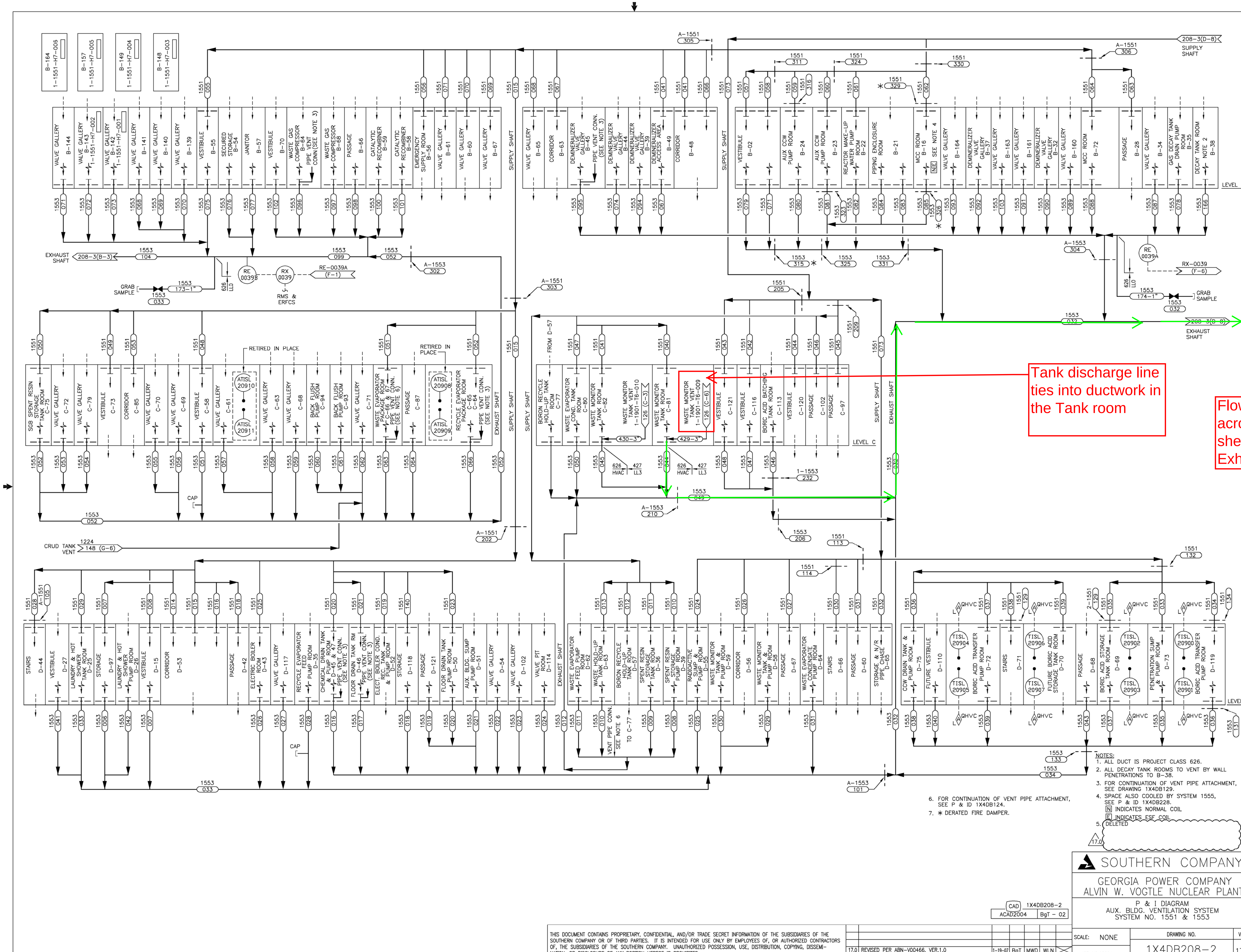
CAD 1X4DB208-1 ACAD2002 PVK - 03

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
AUXILIARY BUILDING OUTSIDE AIR SUPPLY AND NORMAL HVAC SYSTEM (SHEET 1 OF 3)
SYSTEM NO. 1551 & 1553

SCALE: NONE
JOB NO.10604
DRAWING NO. 1X4DB208-1
VER. 27.0
SIZE E 34X44
1 DRAWING CATEGORY: CRITICAL



Tank discharge line ties into ductwork in the Tank room
Flow path goes across several sheets to the Exhaust units
EXHAUST SHAFT
SUPPLY SHAFT
LEVEL B
LEVEL C
LEVEL D
NOTES:
1. ALL DUCT IS PROJECT CLASS 626.
2. ALL DECAY TANK ROOMS TO VENT BY WALL PENETRATIONS TO B-38.
3. FOR CONTINUATION OF VENT PIPE ATTACHMENT, SEE DRAWING 1X4DB129.
4. SPACE ALSO COOLED BY SYSTEM 1555, SEE P & ID 1X4DB228.
N INDICATES NORMAL COIL
E INDICATES ESF COIL
5. DELETED
6. FOR CONTINUATION OF VENT PIPE ATTACHMENT, SEE P & ID 1X4DB124.
7. * DERATED FIRE DAMPER.
SOUTHERN COMPANY
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
P & I DIAGRAM
AUX. BLDG. VENTILATION SYSTEM
SYSTEM NO. 1551 & 1553
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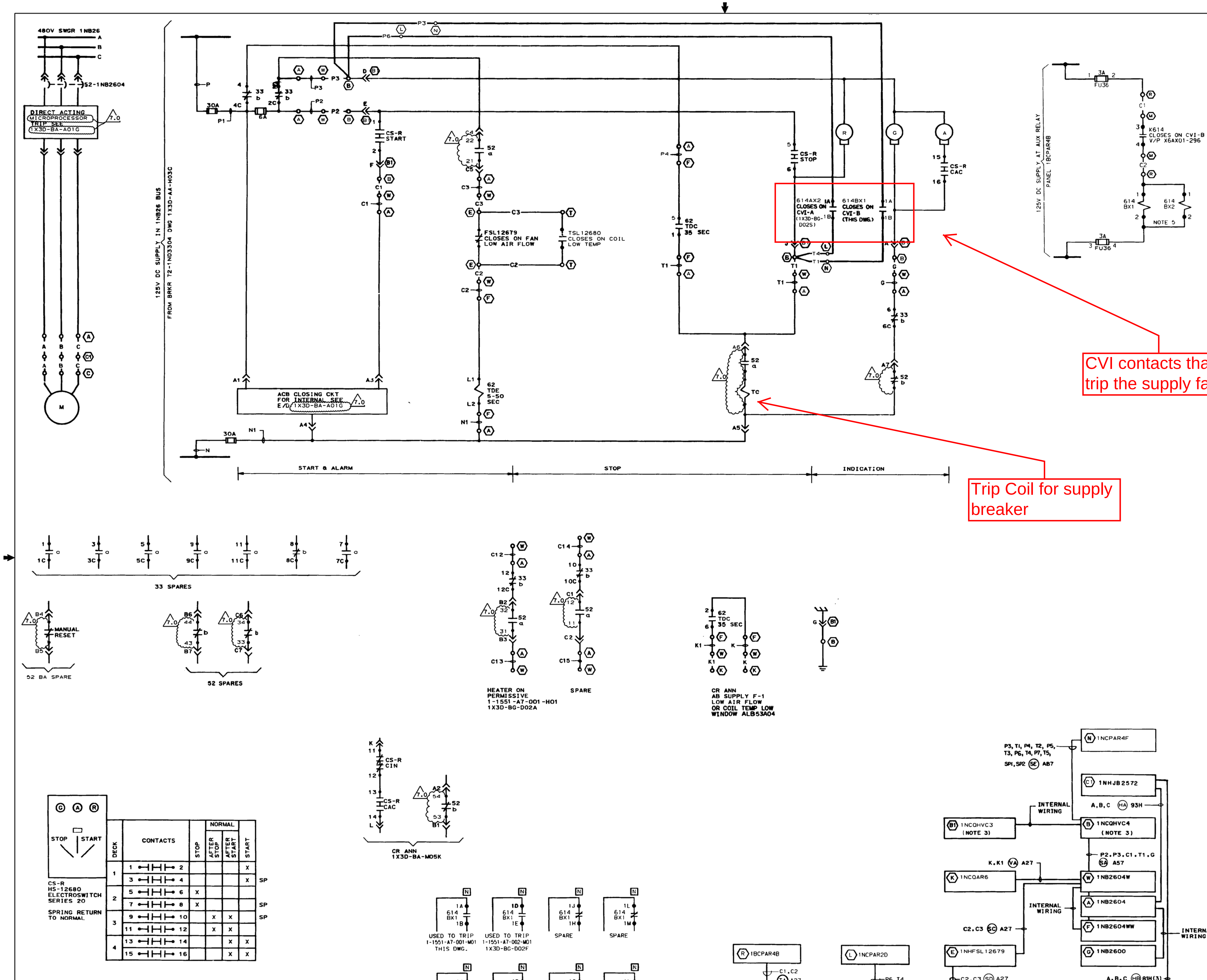
17.0 REVISED PER ABN-V00466, VER.1.0 1-19-07 BgT MWD WLN
NO. VERSIONS DATE DR CHK APPV DTL
SCALE: NONE
JOB NO.10604
DRAWING NO. 1X4DB208-2
VER. 17.0
CAD 1X4DB208-2 ACAD2004 BgT - 02
SIZE E 34X44
DRAWING CATEGORY: CRITICAL



DEVICE	MFR	TYPE	MODEL NO
62	AGA	7000	7012PD
614BX1	P&B	MDR	MDR137-8
614BX2	P&B	MDR	MDR137-8

REFERENCE DWGS		
NUMBER	DESCRIPTION	REV
1X4DB208-1	P & I DIAGRAM	
1X3D-AA-E28A	ONE LINE DIAGRAM	
1X5DN050-1	LOGIC DIAGRAM	
N/A	BECHTEL V/P	

NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DRAWINGS LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. HS-12680 IS LOCATED AT B1. INTERNAL CONNECTION BETWEEN B1 AND B IS BY VENDOR. THE TERMINAL BLOCK IS LOCATED AT B.
4. CONDUCTORS P6 & T4 ARE PART OF CABLE INB2004SE, SHOWN ON DWG. 1X3D-BG-D02S.
5. RELAYS 614BX1 & 614BX2 ARE ISOLATION DEVICES. COIL CKTS. OF THESE RELAYS ARE CLASS 1E (TRAIN B) AND OUTPUT CONTACTS CKTS ARE NON CLASS 1E (NON TRAIN).

NUCLEAR SAFETY RELATED B

AUX BLDG NORMAL AC UNIT FAN MOTOR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

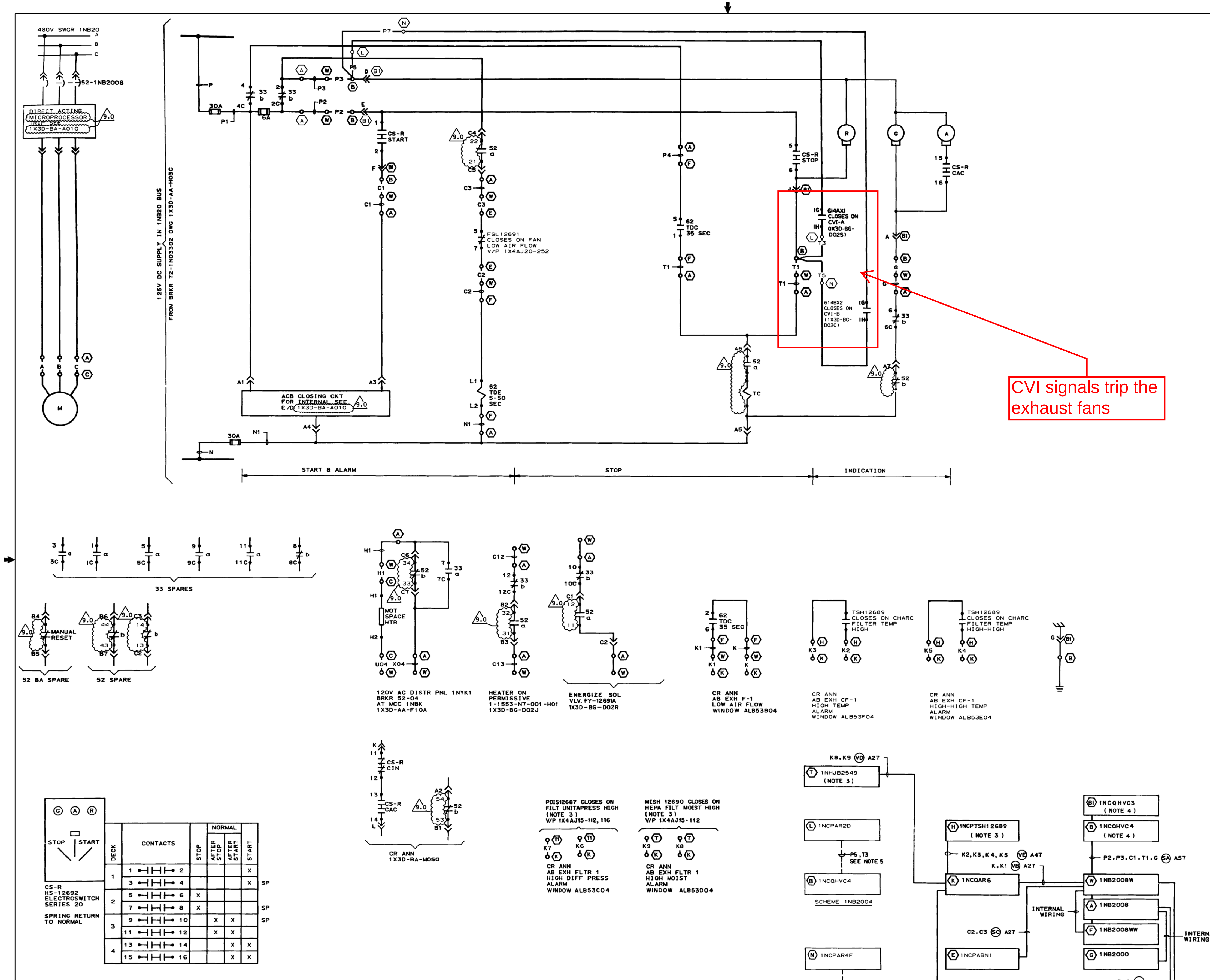
ELEMENTARY DIAGRAM
AUX BLDG OUTSIDE AIR SUPPLY
AND NORMAL HVAC SYSTEM
1-1551-A7-001-M01

NO.	VERSIONS	DATE	DR	CHK	APPV
7.0	REVISED PER ABN C053319401E027, VER.1.0	02/28/12	TSL	GDM	JMR

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-BG-D02C	7.0

SIZE E 34X44

DWG CATEGORY: CRITICAL



CONTACTS		STOP	AFTER STOP	AFTER START	START	
DECK 1	1 — 2				X	
	3 — 4				X	SP
DECK 2	5 — 6	X				
	7 — 8	X				SP
DECK 3	9 — 10		X	X		SP
	11 — 12		X	X		
DECK 4	13 — 14			X	X	
	15 — 16			X	X	

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	DESCRIPTION	REV
62	AGA	7000	7012PD	1X4DB208-1	P & I DIAGRAM	—
				1X3D-AA-E21A	ONE LINE DIAGRAM	—
				1X5DN026-1	LOGIC DIAGRAM	—
				AX4AJ15-118	BECHTEL V/P	—

NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DRAWINGS LISTED ON 1X3D-AA-A00B
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. DEVICES 1NHMISH12690, 1NHMT12690B & 1NHME12690B ARE ASSOCIATED WITH 1NHJB2549. DEVICE 1NHPDIS12687 IS ASSOCIATED WITH 1NHJB2603 DEVICE 1NHTE12689 IS ASSOCIATED WITH 1NCPTSH12689 ALL CABLES FOR THESE DEVICES ARE PRE-WIRED BY THE VENDOR SEE V/P 1X4AJ15-112, 116 & 136 JB'S 2549 & 2603 ARE VENDOR SUPPLIED
4. HS-12692 IS LOCATED AT B1 INTERNAL CONNECTION BETWEEN B1 AND B IS BY RELIANCE. THE TB IS LOCATED AT B
5. CONDUCTORS P5 & T3 ARE PART OF CABLE INB2004SE SHOWN ON DWG. IX3D-BG-D02S.
6. CONDUCTORS P7 & T5 ARE PART OF CABLE INB2604SE SHOWN ON DWG. IX3D-BG-D02C.

AUX BLDG CONT EXH UNIT FAN MOTOR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
AUX BLDG OUTSIDE AIR SUPPLY
AND NORMAL HVAC SYSTEM
1-1553-N7-001-M01

NO.	VERSIONS	DATE	DR	CHK	APPV
9.0	REVISED PER ABN C053319401E013, VER. 1.0	10/19/10	JLO	GDM	JMR

SCALE: NONE	DRAWING NO.	VER.
JOB NO. 10604	1X3D-BG-D02L	9.0

SIZE E 34X44

DWG. CATEGORY: CRITICAL

1. 059K4.19 001/LOIT/RO/M/F 3.2/3.4/059K4.19/LO-TA-28018///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- Reactor trips due to a feedwater transient.
- ALB13-E04 FWI SI OR P14 SG HI-HI LVL is lit.

Which one of the following completes the following statement?

The Main Feedwater System will respond by __(1)__,

and

in order to restore Main Feedwater capability, the reactor trip breakers __(2)__ required to be cycled.

A. (1) closing all FWI valves ONLY

(2) are

B. (1) closing all FWI valves ONLY

(2) are NOT

C.✓ (1) closing all FWI valves and tripping both MFPTs

(2) are

D. (1) closing all FWI valves and tripping both MFPTs

(2) are NOT

059 Main Feedwater (MFW) System

K4.19 Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following:

- Automatic feedwater isolation of MFW

<u>K/A MATCH ANALYSIS</u>

The question tests the candidate's knowledge of the P-14 and P-4 interlocks associated with automatic FWI and how the isolation is reset.

<u>EXPLANATION OF REQUIRED KNOWLEDGE</u>

Per SSPS Logic 1X6AA02-00237, P-14 will initiate a full Feed Water Isolation (FWI) - Main Feed Pumps trip and all main and bypass feed water isolation and regulating valves close.

P-4 has two interlocks associated with FWI isolation. First, P-4 in conjunction with Lo Tavg (<564F) closes all FW valves. Second, P-4 seals in an SI or P-14 signal if present.

Resetting FWI isolation requires that the SI signal be cleared or blocked and the P-14 signal must be cleared. Reactor trip breakers must be cycled to break the seal in. Lo Tavg FWI must be reset. EOP 19231-C step 22 depicts this sequence well.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. With P-14 present, MFPs will trip and all FWI valves will close. However, FWI isolation from P-4 and Lo Tavg will only close the FWI valves. A candidate without detailed knowledge of this circuit and interlocks may not realize P-14 also trips the MFPs and attribute pump trips only to SI.

The second part is correct. P-4 seals in the P-14 FWI isolation signal. Once P-14 is cleared, RTBs must be cycled to break the seal-in.

B. Correct. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. P-4 seals in the P-14 FWI isolation signal. Once P-14 is cleared, RTBs must be cycled to break the seal-in. However, if the candidate does not realize that P-14 is sealed in by P-4, then it would be reasonable to believe that resetting FWI would be the only action required. Furthermore, if the reactor were at 4% power and a P-14 were to occur, cycling the RTBs would not be necessary since a manual reactor trip would not be required due to being on AFW. In this situation, clearing P-14 and resetting FWI would be all that was necessary.

C. Incorrect. Plausible. The first part is correct. With P-14 present, MFPs will trip and all FWI valves will close.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the wecond part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	059K4.19
Importance Rating:	3.2 / 3.4

Technical Reference: 1X6AA02-00237, Rev 10.0
EOP 19231-C, Rev 34.0, page 14

References provided: None

Learning Objective:

LO-PP-28103-05	List all ESF actuation signals with applicable set points, coincidences, permissives, blocks, and discuss the systems response to each ESF actuation signal.
LO-PP-16101-07	State the set point, coincidence, and the purpose for FWI on low Tavg.
LO-PP-20101-06	Given a set of plant conditions, analyze the data to determine if a secondary heat sink is properly established and any actions you would take to establish the heat sink if necessary.
LO-TA-37051	Respond to a Loss of Secondary Heat Sink per 19231-C
LO-TA-28018	Reset Feedwater Isolation

Question origin: BANK - HL16 059K4.19

Cognitive Level: M/F

10 CFR Part 55 Content: 41.7

Comments:

You have completed the test!

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

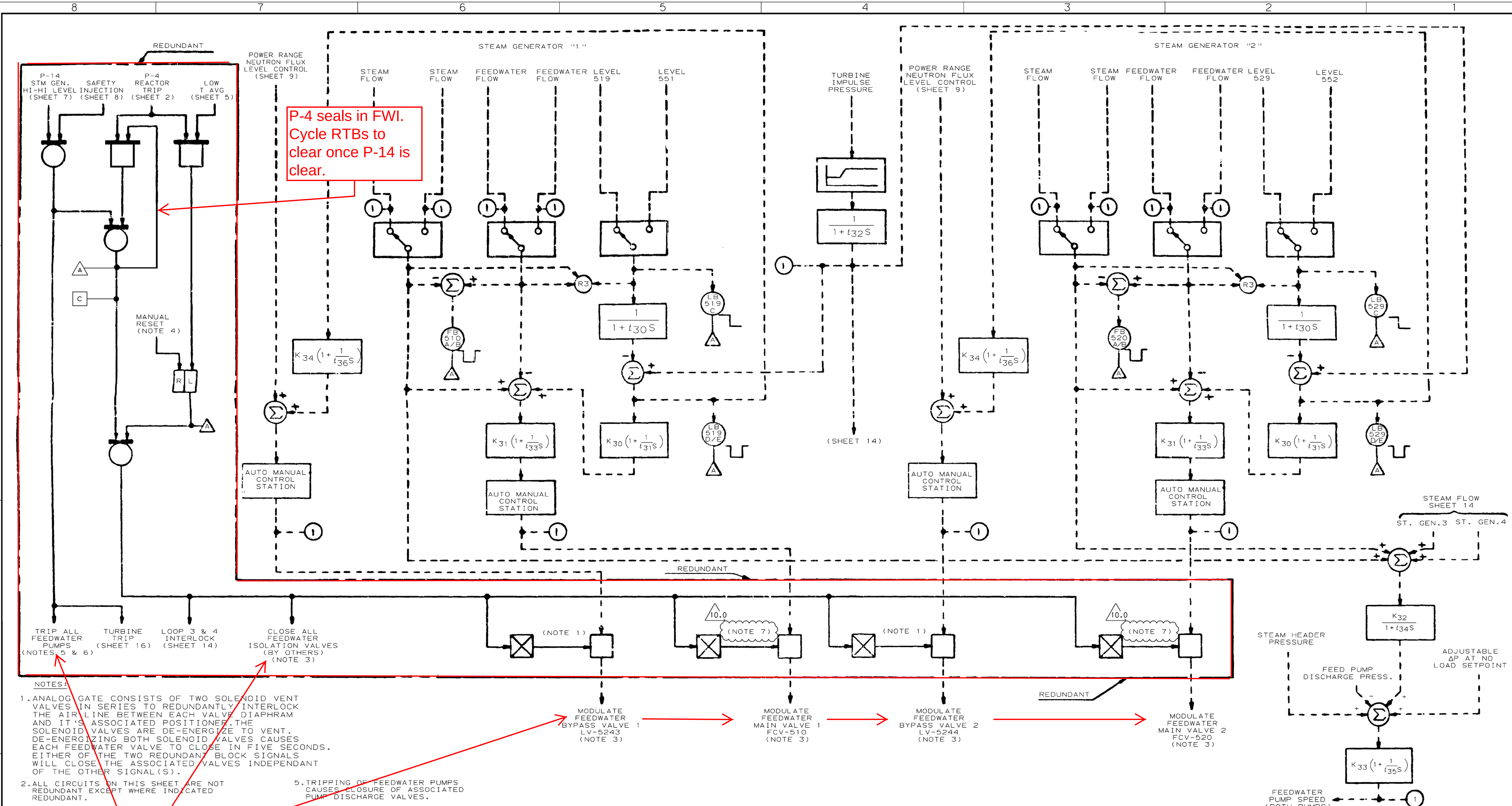
CAUTIONS

- If offsite power is lost after SI reset, action is required to restart the following ESF equipment if plant conditions require their operation:
- RHR Pumps
- SI Pumps
- Post-LOCA Cavity Purge Units
- Containment Coolers in low speed (Started in high speed on a UV signal).
- ESF Chilled Water Pumps (If CRI is reset).

22. Perform the following:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Reset SI.	a. IF SI will NOT reset, THEN initiate Attachment 5.
b. Close RTBs.	
c. Reset FW Isolation.	
d. Energize Stub Busses by performing the following as necessary:	

NB01	NB10
1) Open breaker NB01-01 2) Close breaker AA02-22 3) Close breaker NB01-01	1) Open breaker NB10-01 2) Close breaker BA03-18 3) Close breaker NB10-01



NOTES:

1. ANALOG GATE CONSISTS OF TWO SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAM AND IT'S ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZE TO VENT. DE-ENERGIZING BOTH SOLENOID VALVES CAUSES EACH FEEDWATER VALVE TO CLOSE IN FIVE SECONDS. EITHER OF THE TWO REDUNDANT BLOCK SIGNALS WILL CLOSE THE ASSOCIATED VALVES INDEPENDANT OF THE OTHER SIGNAL(S).
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT EXCEPT WHERE INDICATED REDUNDANT.
3. OPEN/SHUT INDICATION FOR EACH FEEDWATER VALVE IN CONTROL ROOM.
4. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
5. TRIPPING OF FEEDWATER PUMPS CAUSES CLOSURE OF ASSOCIATED PUMP DISCHARGE VALVES.
6. THE FEEDWATER PUMPS AND PUMP DISCHARGE VALVES ARE SUPPLIED BY OTHERS.
7. ANALOG GATE CONSISTS OF TWO SOLENOID VALVES. THE SOLENOID VALVES ARE DE-ENERGIZE TO VENT. DE-ENERGIZING BOTH SOLENOID VALVES CAUSES EACH FEEDWATER VALVE TO CLOSE IN FIVE SECONDS.

10.0

CAD MICROSTATION TSL/01

1X6AA02-00237.DGN

THIS DWG. IS REFERENCED IN VENDOR MANUAL: N/A
TAB/SECT. N/A
PAGE N/A
FIGURE N/A

VERSION 10.0 DATE 01/11/08
REVISED BY SNC PER ABN-V00442, VER.3.0
SEE MICROFILM FOR PREVIOUS REV. SIGNATURES.

BY	CHK'D	APPR.1	APPR.2
TSL	RBH	JMR	

Southern Nuclear Operating Company, Inc.
FOR
VOGTLE ELECTRIC GENERATING PLANT UNIT NO. 1

TITLE: FUNCTIONAL DIAGRAM FEEDWATER CONTROL & ISOLATION

VENDOR: WESTINGHOUSE P.O.#: PAV-00370
DRAWING NO. 1X6AA02-00237
SIZE D

1. 061AA2.03 001/LOIT/RO/M/F 3.0/3.3/061AA2.03/LO-TA-32007///072A4.01

Radiation monitors are as follows:

- 1RE-002, Containment Low Range Area Monitor
- 1RE-003, Containment Low Range Area Monitor
- 1RE-2565, Containment Vent Monitor

Initial conditions:

- Unit 1 is in Mode 6.
- Core offload is in progress.

Current condition:

- ALB06-E01 CNMT VENT ISO ACTUATION is received.

Which one of the following completes the following statement?

A HIGH radiation level of 15 mrem/hour has been exceeded on __(1)__,

and

the HIGH radiation condition __(2)__ latch in on the SRDC panel requiring manual reset.

	__(1)__	__(2)__
A.✓	1RE-002/1RE-003	does
B.	1RE-002/1RE-003	does NOT
C.	1RE-2565	does
D.	1RE-2565	does NOT

K/A

061 **ARM System Alarms**

AA2.03 **Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:**

- Setpoints for alert and high alarms.

K/A MATCH ANALYSIS

The question tests the candidate's ability to recall the setpoint of the high alarms for the CVI area rad monitors RE-002 and 003 in comparison to CVI effluent rad monitor RE-2565.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17006-1 for ALB06-E01 CNMT VENT ISO ACTUATION and ARP 17100-1, a CVI can be generated by either rad monitors RE-002 and RE-003 or RE-2565A, B, & C. Per TS 3.3.6 and ARP 17102-1, RE-002 and RE-003 high alarm setpoints are set to 15mR/hr during CORE ALTERATIONS. These two rad monitors are normally set at 50 times background. RE-2565 only reads in uCi/cc and monitors the Containment Purge Exhaust effluent.

Per ARP 17102-1 page 5, a high alarm on the SRDC will latch and require a manual reset. RE-002 and RE003 are located on the SRDC, RE-2565 is not.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per ARP 17102-1, RE-002 and RE-003 high alarms are set to 15mR/hr during CORE ALTERATIONS.

The second part is correct. Per ARP 17102-1, any high alarm on the SRDC latches in and requires a manual reset.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. Per ARP 17102-1, any high alarm on the SRDC latches in and requires a manual reset. However, RE-2565A, B, & C are not on the SRDC. A candidate with insufficient knowledge of rad monitor locations may conclude that RE-002 and RE-003 are also not on the SRDC.

C. Incorrect. Plausible. The first part is incorrect. Per ARP 17102-1, RE-002 and RE-003 high alarms are set to 15mR/hr during CORE ALTERATIONS. However, RE-2565 reads in uCi/cc and does not monitor containment area radiation, but containment effluent. A candidate without sufficient knowledge of rad monitor setpoints could easily confuse RE-002/003 and RE-2565A, B, & C and believe RE-2565 reads in mR/hr.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G2
K/A#	061AA2.03
Importance Rating:	3.0 / 3.3

Technical Reference: TS 3.3.6, Amendment No. 158, page 3.3.6-6
ARP 17006-1, Rev 33.1, pages 44-46
ARP 17100-1, Rev 26.2, pages 54-59
ARP 17102-1, Rev 20.3, pages 5, 20, 21, 33, & 34

References provided: None

Learning Objective:

LO-LP-61208-04 State the various evolutions performed during Post Refueling Operations that are considered core alterations.

LO-LP-39207-01 For any given item in section 3.3 of Tech Specs, be able to:
a. State the LCO.
b. State any one hour or less required actions.

LO-TA-32007 Verify Proper Automatic actions to high radiation alarms

Question origin: BANK - HL18 Audit 072A4.01

Cognitive Level: M/F

10 CFR Part 55 Content: 41.11 / 41.13

Comments:

You have completed the test!

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Radiation	1,2,3,4,6[c]	2[a]	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	
a. Gaseous (RE-2565C)				(b)
b. Particulate (RE-2565A)				(b)
c. Iodine (RE-2565B)				(b)
d. Area Low Range (RE-0002, RE-0003)				≤ 15 mr/h[c] ≤ 50x background[d]
4. Safety Injection[d]	1,2,3,4	Refer to LCO 3.3.2, "ESFAS Instrumentation,"Function 1, for all initiation functions and requirements.		

(a) Containment ventilation radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

(b) Setpoints will not exceed the limits of Specifications 5.5.4.h and 5.5.4.i of the Radioactive Effluent Controls Program.

(c) During CORE ALTERATIONS and movement of irradiated fuel assemblies within containment.

(d) During MODES 1, 2, 3, and 4.

WINDOW E01

CNMT VENT ISO ACTUATION

ORIGIN

See Probable Cause.

SETPOINT

Not Applicable

1.0 **PROBABLE CAUSE**

1. Manual CIA actuation.

NOTE

*BPLB, 2.7, CVI RAD, and the applicable TSLB-4 CNMT HIGH RAD status light will be lit if caused by radiation relay.

2. Any automatic CVI actuation including:

   a. RE-0002, SSPS Input Relay K164, Train A or B.*

   b. RE-0003, SSPS Input Relay K264, Train A or B.*

   c. RE-2565, SSPS Input Relay K711, Train A or B.*

   d. CVI SSPS Slave Relay K746.

WINDOW E01
(Continued)

2.0 **AUTOMATIC ACTIONS**

Automatic isolation of the following valves:

Computer Point	Valve	Description
ZD9044	HV-12975	CNMT AIR RAD MONITOR INL
ZD9046	HV-12976	CNMT AIR RAD MONITOR INL
ZD9048	HV-12977	CNMT AIR RAD MONITOR OUT
ZD9050	HV-12978	CNMT AIR RAD MONITOR OUT
ZD9204	HV-2626A	CNMT PREACCESS PURGE SUPPLY
ZD9208	HV-2627A	CNMT PREACCESS PURGE SUPPLY
ZD9206	HV-2626B	CNMT MINI PURGE SUPPLY
ZD9210	HV-2627B	CNMT MINI PURGE SUPPLY
ZD9212	HV-2628A	CNMT PREACCESS PURGE EXH
ZD9216	HV-2629A	CNMT PREACCESS PURGE EXH
ZD9214	HV-2628B	CNMT MINI PURGE EXH
ZD9218	HV-2629B	CNMT MINI PURGE EXH
ZD9236	HV-2624A	CTB POST LOCA PURGE EXH
ZD9238	HV-2624B	CTB POST LOCA PURGE EXH
ZD9583	HV-12604	AUX BLDG VENT SYS SUPPLY
ZD9587	HV-12605	AUX BLDG VENT SYS RETURN
ZD9589	HV-12606	AUX BLDG VENT SYS RETURN
ZD9585	HV-12607	AUX BLDG VENT SYS SUPPLY
NONE	HV-12596	RECYCLE HOLDUP TANK
NONE	HV-12597	RECYCLE HOLDUP TANK

3.0 **INITIAL OPERATOR ACTIONS**

1. **Check** that valves and dampers align per automatic actions.

2. **Determine** cause of CVI:

   a. **Check** CNMT HIGH RAD trip status lights on TSLB-4,

      16.1 – 1-RE-0002 CNMT HIGH RAD,
      16.2 – 1-RE-0003 CNMT HIGH RAD,
      16.3 – 1-RE-2565 CNMT HIGH RAD,

   b. **check** for high radiation levels on the DRMS Communications Console and the Integrated Plant Computer.

WINDOW E01
(Continued)

3. **Notify** HP and plant management of CVI actuation due to high radiation.

4. Take actions to **mitigate** cause of high radiation.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. IF equipment failure is indicated, **initiate** maintenance.

2. WHEN the cause of the CVI has been corrected and it has been determined that the CVI actuated equipment is ready to return to service, **initiate** CVI recovery action per 11886-1, "Recovery From ESF Actuations."

3. **Refer** to Technical Specification 3.4.15 due to the isolation of the containment atmosphere rad monitor, 1RE-2562.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X6AA02-232, 2X6AX01-305, -383

ORIGIN	SETPOINT	
Moving Paper Airborne Particulate Effluent Monitor	As determined by Chemistry Department	1-RE-2565A (High)

NOTES

- This Moving Paper Particulate Monitor has 1 hour time constant so indicator changes should be slow. ☐
- For other than HIGH conditions see Pages 4 and 5. ☐

1.0 **PROBABLE CAUSE**

High level of radiation from airborne Particulates in Containment Purge Vent.

2.0 **AUTOMATIC ACTIONS**

Initiates Containment Ventilation Isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Initiate** evacuation of Containment IF the alarm is due to unexpected or unexplained radiation increases, OR IF appropriate HP controls are NOT in place for the radiological conditions indicated.

2. IF the alarm is due to expected radiation increases from preplanned evolutions AND appropriate HP controls are in place, THEN **request** HP and Chemistry to investigate the cause of alarm. IF required, **initiate** evacuation of Containment.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NOTE

Exhaust gasses are monitored at the plant vent by 1-RE-12442 A, B and C. ☐

1. **Verify** Containment Ventilation Isolation by observing MLB Lights.

2. **Account** for all personnel in the Containment.

3. **Notify** Health Physics to survey and determine the source of radioactivity.

1-RE-2565A
(Continued)

4. **Check** for increased level of radioactivity indicated on 1-RE-12442A.

5. **Refer** to NMP-EP-110, "Emergency Classification And Implementing Instructions".

6. **Obtain** detector trend data per 13508-1, "Radiation Monitoring Systems".

7. **Monitor** the channel for further changes.

8. IF the cause was a spurious alarm AND WHEN conditions permit, **have** Chemistry reset and return the monitor to normal.

9. IF sampling and analysis determine the channel has malfunctioned:

   a. **Comply** with Technical Specification LCO 3.3.6.

   Critical

   b. **Unlock** CVI BLOCK PANEL 1-1609-P5-CB3 (Equipment Bldg R-117), and **place** 1-HS-13261 to BLOCK,

   ________ Initial

   ________ CV Initial

   c. **Request** Chemistry to investigate and take corrective action.

10. WHEN conditions permit, **reset** CVI per 11886-1, "Recovery From ESF Actuations".

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB213-1, 1X5DX4151

ORIGIN	SETPOINT	
Effluent Iodine Monitor	As determined by Chemistry Department	1-RE-2565B (High)

NOTE

For other than HIGH conditions see Pages 4 and 5. ☐

1.0 **PROBABLE CAUSE**

High concentration of gaseous radioactive iodine in the Containment Purge Vent.

2.0 **AUTOMATIC ACTIONS**

Initiates Containment Ventilation Isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Initiate** evacuation of Containment IF the alarm is due to unexpected or unexplained radiation increases, OR IF appropriate HP controls are NOT in place for the radiological conditions indicated.

2. IF the alarm is due to expected radiation increases from preplanned evolutions AND appropriate HP controls are in place, THEN **request** HP and Chemistry to investigate the cause of alarm. IF required, **initiate** evacuation of Containment.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NOTE

Exhaust gasses are monitored at the plant vent by 1-RE-12442A, B and C. ☐

1. **Verify** Containment Ventilation Isolation by observing MLB Lights.

2. **Account** for all personnel in the containment.

3. **Notify** Health Physics to survey and determine the source of radioactivity.

4. **Check** for increased level of radioactivity indicated on 1-RE-12442B.

1-RE-2565B
(Continued)

5. **Refer** to NMP-EP-110, "Emergency Classification And Implementing Instructions".

6. **Obtain** detector trend data per 13508-1, "Radiation Monitoring Systems".

7. **Monitor** the channel for further changes.

8. IF the cause was a spurious alarm AND WHEN conditions permit, **have** Chemistry reset and return the monitor to normal.

9. IF sampling and analysis determine the channel has malfunctioned:

   a. **Comply** with Technical Specifications LCO 3.3.6.

   Critical
   b. **Unlock** CVI BLOCK PANEL 1-1609-P5-CB3 (Equipment Bldg R-117), and **place** 1-HS-13261 to BLOCK,

   ________ Initial

   ________ CV Initial

   c. **Request** Chemistry to investigate and take corrective action.

10. WHEN conditions permit, **reset** CVI per 11886-1, "Recovery From ESF Actuations".

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB213-1, 1X5DX4151

ORIGIN	**SETPOINT**	
Effluent Radiogas Monitor	As determined by Chemistry Department	1-RE-2565C (High)

> NOTE
>
> For other than HIGH conditions see Pages 4 and 5. ☐

1.0 **PROBABLE CAUSE**

Increase in concentration of radioactive gas in the Containment Purge Vent.

2.0 **AUTOMATIC ACTIONS**

Initiates Containment Ventilation Isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Initiate** evacuation of Containment IF the alarm is due to unexpected or unexplained radiation increases, OR IF appropriate HP controls are NOT in place for the radiological conditions indicated.

2. IF the alarm is due to expected radiation increases from preplanned evolutions AND appropriate HP controls are in place, THEN **request** HP and Chemistry to investigate the cause of alarm. IF required, **initiate** evacuation of Containment.

4.0 **SUBSEQUENT OPERATOR ACTIONS**

> NOTE
>
> Exhaust gasses are monitored at the plant vent by 1-RE-12442A, B and C. ☐

1. **Verify** Containment Ventilation Isolation by observing MLB Lights.

2. **Account** for all personnel in the containment.

3. **Notify** Health Physics to survey and determine the source of radioactivity.

4. **Check** for increased level of radioactivity indicated on 1-RE-12442C.

1-RE-2565C
(Continued)

5. **Refer** to NMP-EP-110, "Emergency Classification And Implementing Instructions".

6. **Obtain** detector trend data per 13508-1, "Radiation Monitoring Systems".

7. **Monitor** the channel for further changes.

8. IF the cause was a spurious alarm AND WHEN conditions permit, **have** Chemistry reset and return the monitor to normal.

9. IF sampling and analysis determine the channel has malfunctioned:

   a. **Comply** with Technical Specifications LCO 3.3.6.

   Critical
   b. **Unlock** CVI BLOCK PANEL 1-1609-P5-CB3 (Equipment Bldg R-117), and **place** 1-HS-13261 to BLOCK,

   ____
   Initial

   ____
   CV Initial

   c. **Request** Chemistry to investigate and take corrective action.

10. WHEN conditions permit, **reset** CVI per 11886-1, "Recovery From ESF Actuations".

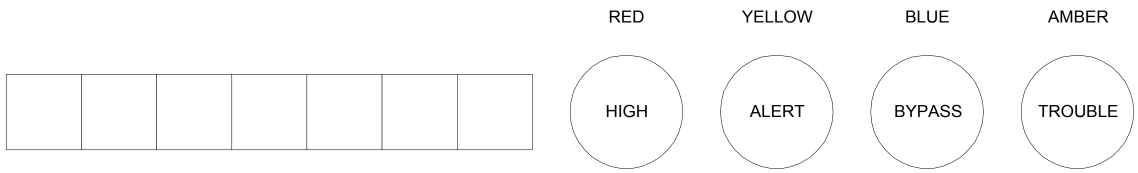
5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB213-1, 1X5DX4151

Each channel on the SRDC has a separate display. Normally each display reads the radiation activity level being monitored in 3 digits and an exponent. Units vary from channel to channel. Each channel has an Alert, High and Equipment Trouble alarm display and an indicator that the SRDC is bypassed:



On detecting a high radiation level, the audible alarm on ALB 05 sounds and the red HIGH indicator lamp on the SRDC channel lights. The alarm is also indicated in the TSC and the Health Physics and Chemistry Labs, by displaying the channel identification number on their CRT in red. The alarm is also displayed at the Communications Console (QRM1). A loud horn and a strobe light may announce the high alarm close to the detector.

Alert, Bypass and Equipment Trouble indications do not sound audible alarms.

For very high radiation levels, the TOP OF SCALE, the EQUIPMENT TROUBLE and the HIGH alarms will all light and the sections of the digital display go to "9999999". This causes the alarm to latch, so it will not automatically clear when the radiation level drops. The TOP OF SCALE must be manually reset at the Channel Display and Control Area (CDCA) on the SRDC.

A high alarm will also latch and require a manual resetting.

The Bypass indicates that the channel has been put in the local control mode at the Data Processing Module (DPM) and for 1-RE-0002 or 1-RE-0003 the Containment Ventilation Isolation Block Switches have been placed in the BLOCK position.

WINDOW CDCA B3

ORIGIN	SETPOINT	
Containment Low Range Area Monitor 1-RE-0002	As determined by Chemistry (15 mR/hr during refueling)	1-RE-0002 (RED LAMP LIT) (HIGH)

NOTE

For other than HIGH conditions see Pages 5 and 6. ☐

1.0 **PROBABLE CAUSE**

NOTES

- During refueling operations indicates a fuel drop accident. ☐
- During power operation indicates possible loss of coolant accident. ☐

High radiation in the Containment Building.

2.0 **AUTOMATIC ACTIONS**

Initiates Containment Building Ventilation Isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Initiate** evacuation of Containment IF the alarm is due to unexpected or unexplained radiation increases, OR IF appropriate HP controls are NOT in place for the radiological conditions indicated.

2. IF the alarm is due to expected radiation increases from preplanned evolutions AND appropriate HP controls are in place, THEN **request** HP and Chemistry to investigate the cause of alarm. If required, **initiate** evacuation of Containment.

WINDOW CDCA B3
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Verify** Containment Ventilation Isolation.

2. If required, **verify** that Containment has been evacuated and all personnel accounted for.

3. **Notify** Chemistry to independently determine radiation level in the Containment.

4. IF the channel has malfunctioned:

   a. **Comply** with Technical Specifications LCO 3.3.6.

   b. **Unlock** panel and **place** 1-HS-13259 on CVI BLOCK PANEL 1-1609-P5-CB1 to BLOCK (Cont. Bldg RB-70).

   c. **Request** Chemistry to investigate and take corrective action.

   d. **Reset** CVI per 11886-1, "Recovery From ESF Actuations."

5. IF the alarm is an actuation resulting from Fuel Handling, **initiate** 18006-C, **"**Fuel Handling Event,**"** as appropriate.

6. IF the channel has not malfunctioned, **initiate** 18004-C, **"**RCS Leakage.**"**

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB213-2, 1X5DS4C02

WINDOW CDCA C3

ORIGIN	**SETPOINT**	
Containment Low Range Area Monitor 1-RE-0003	As determined by Chemistry (15 mR/hr during refueling)	1-RE-0003 (RED LAMP LIT) (HIGH)

NOTE

For other than HIGH conditions see Pages 5 and 6. ☐

1.0 **PROBABLE CAUSE**

NOTES

- During refueling operations indicates a fuel drop accident. ☐
- During power operation indicates possible loss of coolant accident. ☐

High radiation in the Containment Building.

2.0 **AUTOMATIC ACTIONS**

Initiates Containment Building Ventilation isolation.

3.0 **INITIAL OPERATOR ACTIONS**

1. **Initiate** evacuation of Containment IF the alarm is due to unexpected or unexplained radiation increases, OR IF appropriate HP controls are NOT in place for the radiological conditions indicated.

2. IF the alarm is due to expected radiation increases from preplanned evolutions AND appropriate HP controls are in place, THEN **request** HP and Chemistry to investigate the cause of alarm. If required, **initiate** evacuation of Containment.

WINDOW CDCA C3
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Verify** Containment Ventilation Isolation.

2. If required, **verify** that the Containment has been evacuated and all personnel accounted for.

3. **Notify** Chemistry to independently determine radiation level in the Containment.

4. IF the channel has malfunctioned:

   a. **Comply** with Technical Specifications LCO 3.3.6.

   b. **Unlock** panel and **place** 1-HS-13260 on CVI BLOCK PANEL 1-1609-P5-CB2 to BLOCK (Cont. Bldg RB-38).

   c. **Request** Chemistry to investigate and take corrective action.

   d. **Reset** CVI per 11886-1, "Recovery From ESF Actuations."

5. IF the alarm is an actuation resulting from Fuel Handling, **initiate** 18006-C, "Fuel Handling Event," as appropriate.

6. IF the channel has not malfunctioned, **initiate** 18004-C, "RCS Leakage."

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB213-2, 1X5DS4A02

1. 061K2.01 001/LOIT AND LOCT/RO/M/F 3.2/3.3/061K2.01/LO-PP-20101-09///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- An electrical fault results in the following indications:
  - 1HV-5106, TDAFW Steam Admission Valve, red and green handswitch lights are out.
  - 1FV-5132 and 5134, MDAFW SGs #2 and 3 Discharge Throttle Valves, red, green, and white lights are out.

Which one of the following completes the following statement?

A loss of Train 'C' __(1)__,

and

a loss of Train 'B' __(2)__ power have occurred.

	__(1)__	__(2)__
A.✓	125 VDC	480 VAC
B.	125 VDC	125 VDC
C.	480 VAC	480 VAC
D.	480 VAC	125 VDC

K/A

061 **Auxiliary / Emergency Feedwater (AFW) System**

K2.01 **Knowledge of bus power supplies to the following:**

- AFW system MOVs

K/A MATCH ANALYSIS

The question tests the candidates knowledge of the the AFW system MOV bus power supplies by requiring the candidate to select the specific voltage supplied to two different MOVs - one associated with TDAFW and the other with MDAFW.

EXPLANATION OF REQUIRED KNOWLEDGE

AFW is separated into 3 trains. The two motor driven trains are supplied from its train 4160VAC SWGR and the MOV from its train 480VAC MCC. The TDAFW train is steam driven off loops 1&2. The steam supply isolations from each loop are A or B train powered 125VDC. All other MOVs in the TDAFW system are train 'C' 125VDC powered.

The absence of light indication on the individual MOV handswitches denotes a loss of control power. The control power for the MDAFW pump discharge MOVs comes from a 480VAC/120VAC control power transformer inside the MCC bucket. (Ref ELEMENTARY 1X3D-BC-F08A&B) The control power for the TDAFW MOVs comes from the same 125VDC power that feeds the MOV itself inside the bucket. (Ref ELEMENTARY 1X3D-BC-F02A)

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. 1HV-5106 is powered from 125VDC MCC 1CD1M.

The second part is correct. 1FV-5132 is powered from 480VAC MCC 1BBB.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. 1FV-5132 is powered from 480VAC MCC 1BBB. However, the train 'B' steam supply from loop 1 to the TDAFW pump, 1HV-3009 is powered from 125VDC MCC 1BD1M. It is reasonable for a candidate without specific knowledge of bus power supplies to assume that all the AFW MOVs are powered from the associated train 125VDC.

C. Incorrect. Plausible. The first part is incorrect. 1HV-5106 is powered from 125VDC MCC 1CD1M. However, the vast majority of MOVs are 480VAC powered and 19100-C step 4 RNO directs the operator to locally operate HV5106 using an attachment in AOP 18034-1/2. Manual operation is a sign-off for both licensed and non-licensed operators. It is reasonable for an operator without specific knowledge of bus power supplies to assume that this guidance exists due to a loss of 480VAC power to HV5106 during a Loss of All AC. Additionally, RHR loop suction valves HV8701B and HV8702A are 480VAC MOVs that are powered from a 125VDC/480VAC inverter. It is reasonable for an operator to believe that AFW has a similar arrangement.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	061K2.01
Importance Rating:	3.2 / 3.3

Technical Reference:	EOP 19100-C, Rev 38.1, page 8 AOP 18034-1, Rev 13.1, page 77 & 78 ELEMENTARY 1X3D-BC-F02A, Rev 11.0 ELEMENTARY 1X3D-BC-F08A, Rev 9.0 ELEMENTARY 1X3D-BC-F08B, Rev 10.0 P&ID 1X4DB161-2, Rev 28.0 P&ID 1X4DB161-3, Rev 42.0
References provided:	None
Learning Objective:	LO-PP-20101-09 Determine the impact to AFW system operation and the overall integrated plant operations to the following types of power supply failures: a. U/V condition on either AA02 or BA03 with the bus being re-energized from the EDG while at 100% power. b. U/V condition on either AA02 or BA03 with the bus remaining de-energized while at 100% power. c. Loss of a 120 VAC 1E vital instrument bus. d. Loss of a 125 VDC 1E bus e. Loss of All AC Power LO-TA-20009 TDAFW Pump Local Manual Control using HV-5106 and SOP 13610-1/2
Question origin:	BANK - HL16 NRC Question # 061K2.01
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.7

You have completed the test!

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. Verify AFW flow - GREATER THAN 570 GPM.	4. Perform the following: a. Verify TDAFW Pump is running: • PDIC-5180 - INCREASED/ MAX DEMAND • HV-5106 TDAFW - OPEN • HV-3009 LP-1 MS SPLY TO AUX FW TD PMP-1 - OPEN -OR- • HV-3019 LP-2 MS SPLY TO AUX FW TD PMP-1 - OPEN b. IF TDAFW Pump can NOT be operated normally due to governor or DC power failure, THEN dispatch Operator to attempt local manual control by initiating 18034-1, 18034-2 Loss of Class 1E 125V DC Power. c. Verify AFW Throttle Valves open.
5. Trip all RCPs.	
6. Trip the NCP.	
7. Initiate the following: • Continuous Action Page. • NMP-EP-110 Emergency Classification Determination and Initial Action.	

ATTACHMENT C2 Sheet 1 of 2

TURBINE DRIVEN AFW PUMP LOCAL MANUAL CONTROL WITHOUT DC POWER

NOTE

This attachment gives instructions to operate the TDAFW Pump if DC bus 1CD1 is de-energized. In this case the TDAFW Pump governor valve will fail full open. Steam to the turbine must be manually throttled to prevent overspeed of the turbine and overpressurization of the piping. Some of the local instrumentation including turbine speed and pump discharge pressure will be unavailable.

CAUTION

This attachment shall be used only with the permission of the SS.

1. Establish communications between the Main Control Room and the TDAFW Pump Room.

2. Locally verify closed TDAFW Pump Steam Supply 1-HV-5106.

3. Locally verify open the TDAFW Pump Trip & Throttle valve 1-PV-15129.

NOTE

Without power there is no direct indication of TDAFW pump speed. Miniflow indicator 1-FI-15100 should be used to ascertain TDAFW Pump performance. 140 gpm recirculation flow corresponds to a speed slightly less than normal speed of 4230 rpm with AFW throttle valves shut. If throttle valves are opened, 140 gpm corresponds to a speed above the normal speed of 4230 rpm but still less than the overspeed setpoint of 4830 rpm.

4. Throttle open TDAFW Pump Steam Supply 1-HV-5106 while observing 1-FI-15100. Continue to open valve until approximately 140 gpm is observed on 1-FI-15100.

°

ATTACHMENT C2 Sheet 2 of 2

TURBINE DRIVEN AFW PUMP LOCAL MANUAL CONTROL WITHOUT DC POWER

<u>NOTE</u>

If TDAFW Pump trips, it should be reset using 13610, AUXILIARY FEEDWATER SYSTEM.

5. Adjust steam supply valve position as necessary to control pump speed and flow:

   a. Speed indication (strobe) available - Throttle steam flow to limit speed to 4230 rpm.

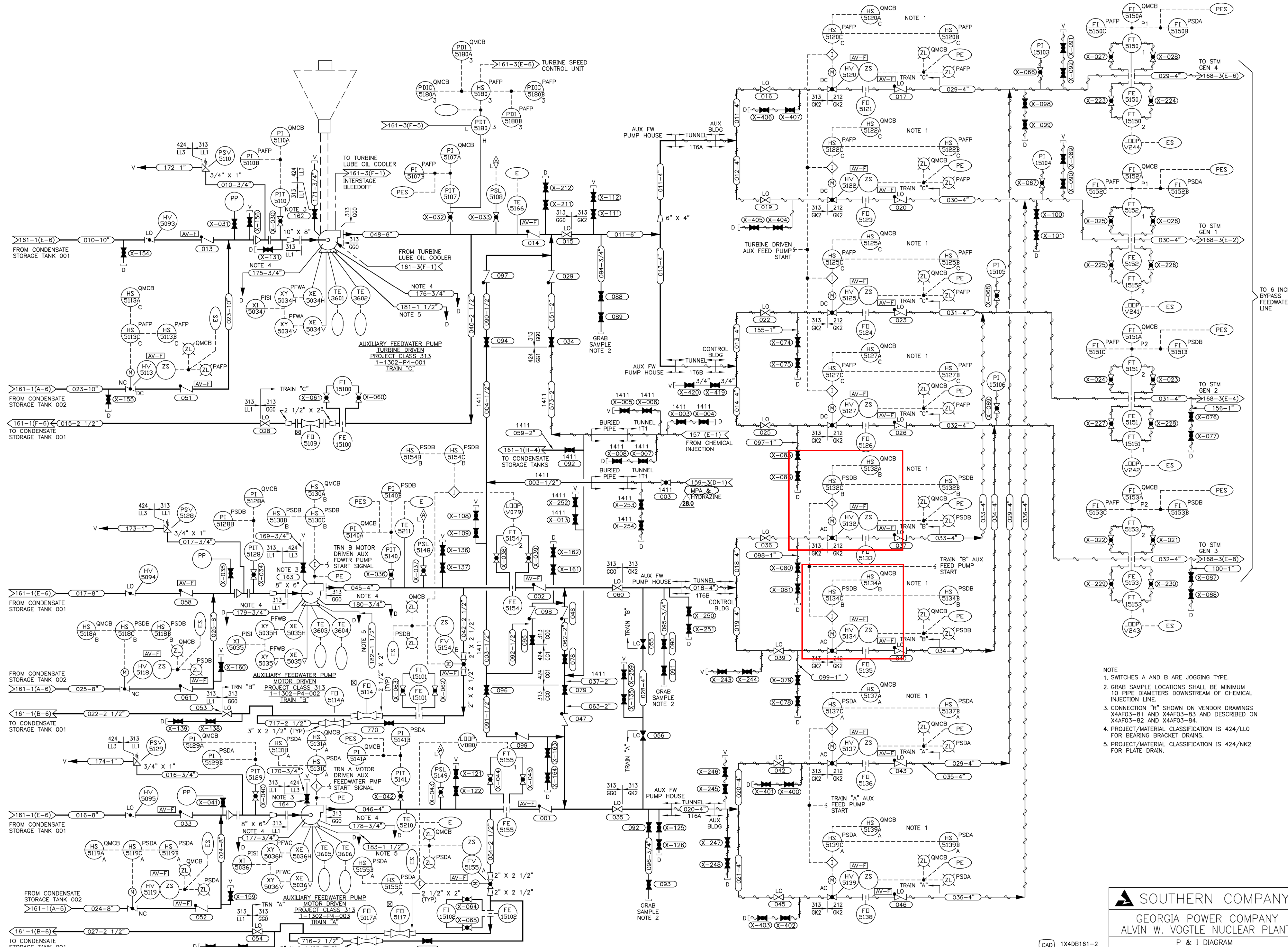
   b. Speed indication not available - Limit recirculation flow indicated on 1-FI-15100 to 140 gpm or discharge pressure to 1700 psig.

6. When operation of the TDAFW pump is no longer required:

   a. Trip pump using the Manual Overspeed Trip Mechanism Thumb Lever.

   b. Restore pump to standby per 13610, AUXILIARY FEEDWATER SYSTEM.

END OF ATTACHMENT C2



NOTE

1. SWITCHES A AND B ARE JOGGING TYPE.
2. GRAB SAMPLE LOCATIONS SHALL BE MINIMUM 10 PIPE DIAMETERS DOWNSTREAM OF CHEMICAL INJECTION LINE.
3. CONNECTION "R" SHOWN ON VENDOR DRAWINGS X4AF03-81 AND X4AF03-83 AND DESCRIBED ON X4AF03-82 AND X4AF03-84.
4. PROJECT/MATERIAL CLASSIFICATION IS 424/LLO FOR BEARING BRACKET DRAINS.
5. PROJECT/MATERIAL CLASSIFICATION IS 424/NK2 FOR PLATE DRAIN.

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
AUXILIARY FEEDWATER SYSTEM
SYSTEM NO. 1302

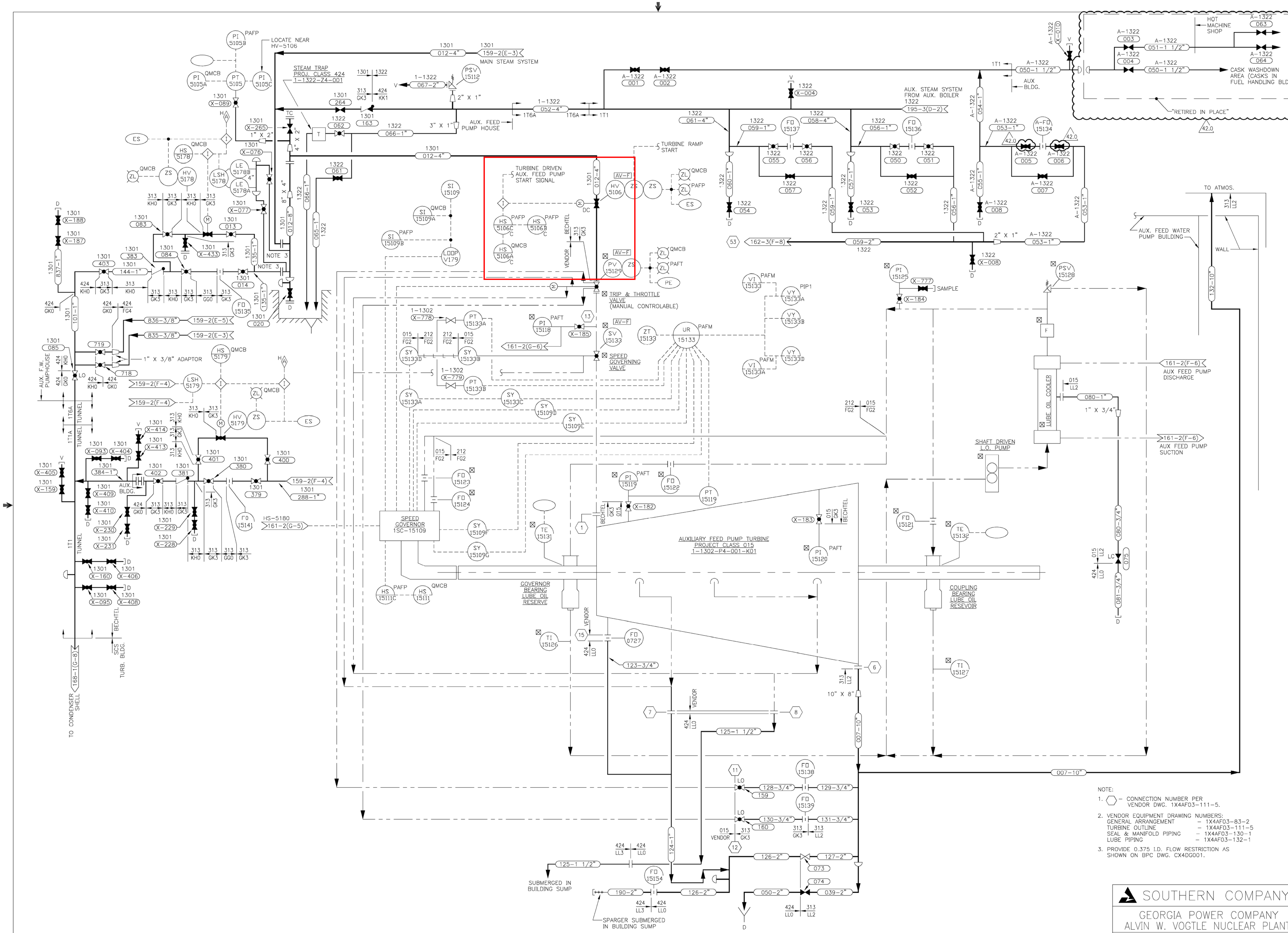
CAD 1X4DB161-2
ACAD2002 KB - 01

NO.		DATE	DR	CHK	APPV	DTL
28.0	REVISED PER ABN 1039001101M058, VER. 1.0	8-26-08	KB	JWM	JMR	X

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB161-2	28.0

SIZE E 34X44

1 DRAWING CATEGORY: CRITICAL



NOTE:

1. ⬡ – CONNECTION NUMBER PER VENDOR DWG. 1X4AF03-111-5.
2. VENDOR EQUIPMENT DRAWING NUMBERS:

GENERAL ARRANGEMENT	– 1X4AF03-83-2
TURBINE OUTLINE	– 1X4AF03-111-5
SEAL & MANIFOLD PIPING	– 1X4AF03-130-1
LUBE PIPING	– 1X4AF03-132-1

3. PROVIDE 0.375 I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG. CX4DG001.

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
AUXILIARY FEEDWATER PUMP SYSTEM
(AUX FEEDWATER PUMP TURBINE DRIVER)
SYSTEM 1302

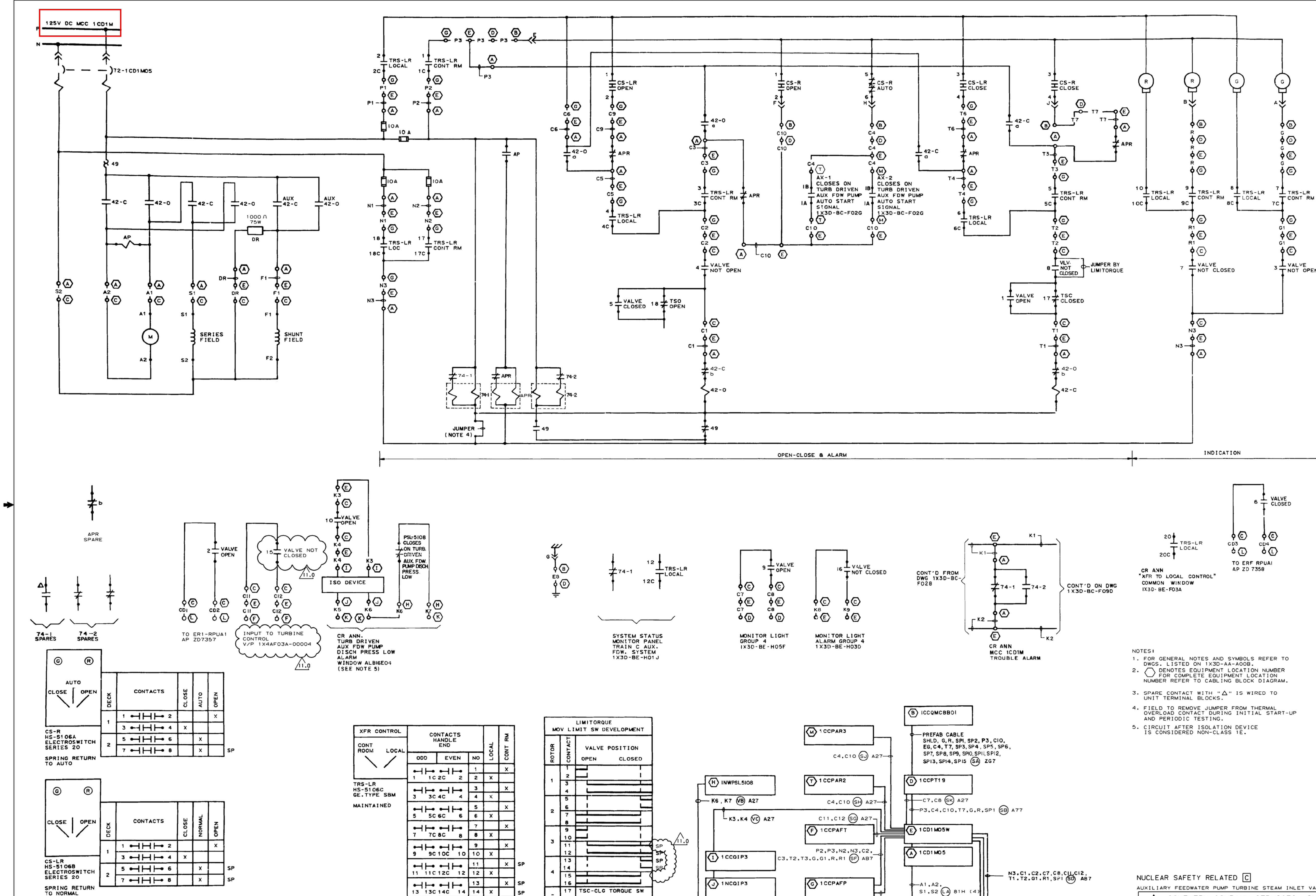
CAD 1X4DB161-3 | ACAD2004 | KB - 03

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
42.0	REV. PER ABN C100867701M006, VER. 1.0.	07/16/12	KB	RBH	JMR	X

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X4DB161-3 | VER. 42.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DWGS. LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH "△" IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP AND PERIODIC TESTING.
5. CIRCUIT AFTER ISOLATION DEVICE IS CONSIDERED NON-CLASS 1E.

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	REFERENCE DWGS DESCRIPTION	REFERENCE DWGS REV
74-1	CH	D26M	D26MRD12A1	1X4DB161-3	P & I DIAGRAM	—
74-2	CH	D26M	D26MRD12A1	1X3D-AA-H04A	ONE LINE DIAGRAM	—
AP	CH	900	10-2907	1X5DN120-2	LOGIC DIAGRAM	—
APR	CH	D26M	D26MRD04A1	AX4AR01-617	BECHTEL V/P	—
DR	CH		11003H44B22			

NO.	VERSIONS	DATE	DR	CHK	APPV
11.0	REV. PER ABN 1019003501E004, VER.1.0	4-4-05	JLO	TSL	DEW

NUCLEAR SAFETY RELATED C Q

AUXILIARY FEEDWATER PUMP TURBINE STEAM INLET VALVE

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

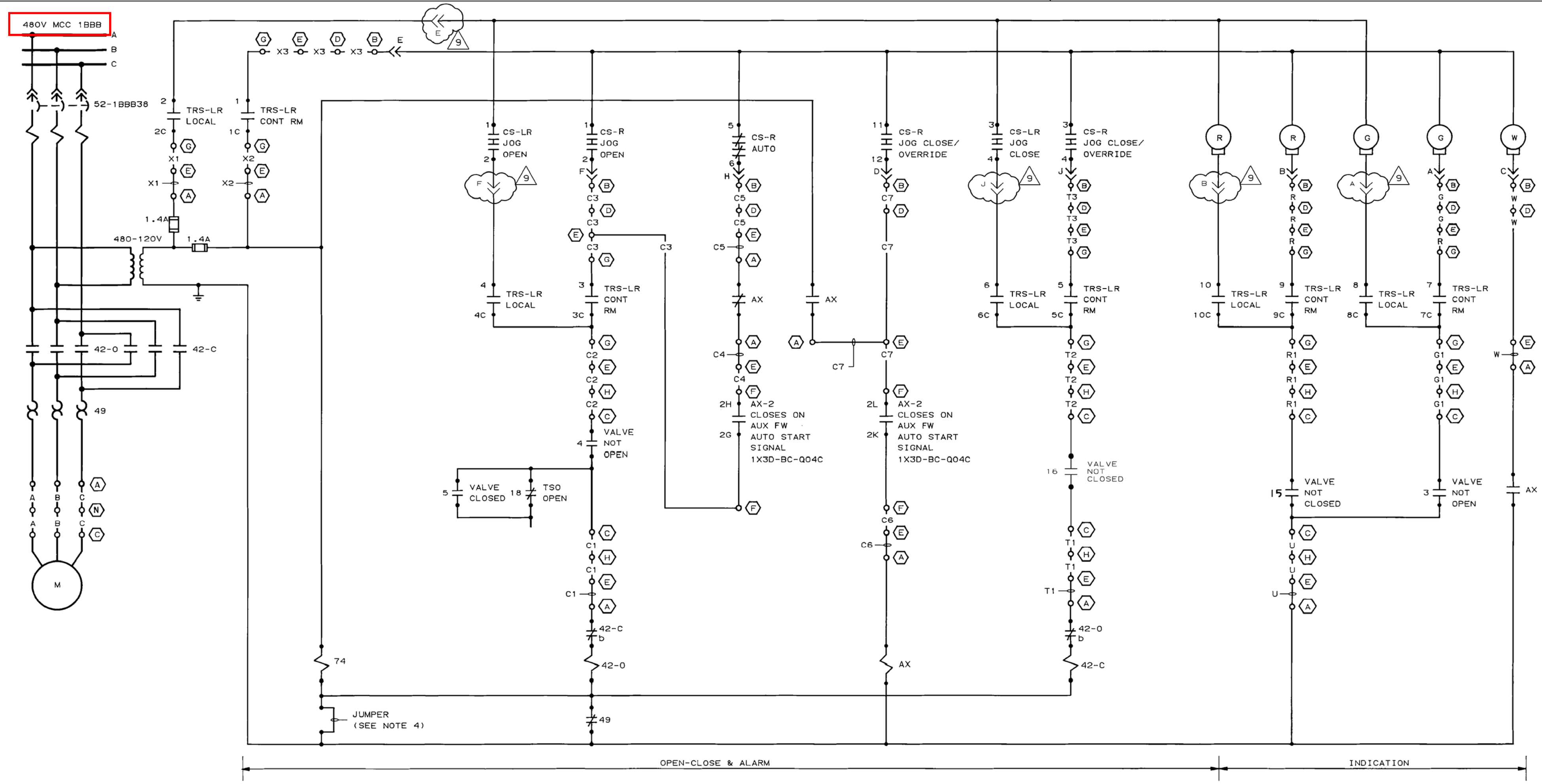
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

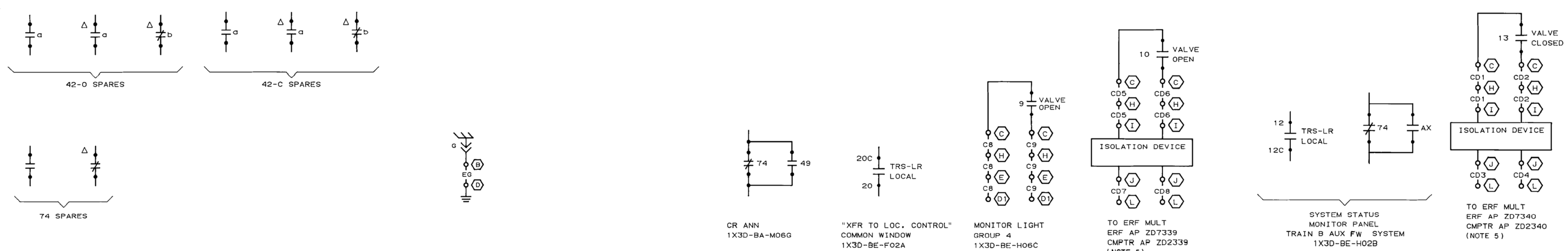
ELEMENTARY DIAGRAM
AUX FEEDWATER SYSTEM
1HV-5106

SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X3D-BC-F02A | VER. 11.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL





CS-R
HS-5132A
ELECTROSWITCH,
SERIES 20
SPRING RETURN
TO AUTO

Positions: JOG CLOSE/OVERRIDE, AUTO, JOG OPEN

DECK	CONTACTS	JOG CLOSE/ OVERRIDE	AUTO	JOG OPEN	
1	1 – 2			X	
1	3 – 4	X			
2	5 – 6		X		
2	7 – 8		X		SP
3	9 – 10			X	SP
3	11 – 12	X			
4	13 – 14		X		SP
4	15 – 16		X		SP

CS-LR
HS-5132B
ELECTROSWITCH,
SERIES 20
SPRING RETURN
TO NORMAL

Positions: JOG CLOSE, JOG OPEN

DECK	CONTACTS	JOG CLOSE	NORMAL	JOG OPEN	
1	1 – 2			X	
1	3 – 4	X			
2	5 – 6		X		SP
2	7 – 8		X		SP

XFR CONTROL

CONT RM / LOCAL

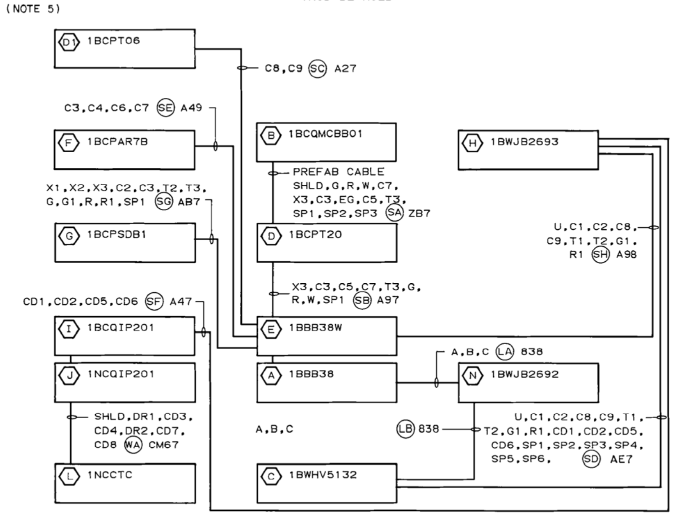
TRS-LR
HS-5132C
GE, TYPE SBM
MAINTAINED

CONTACTS HANDLE END ODD	CONTACTS HANDLE END EVEN	NO	LOCAL	CONT RM	
1 1C	2C 2	1		X	
		2	X		
3 3C	4C 4	3		X	
		4	X		
5 5C	6C 6	5		X	
		6	X		
7 7C	8C 8	7		X	
		8	X		
9 9C	10C 10	9		X	
		10	X		
11 11C	12C 12	11		X	SP
		12	X		
13 13C	14C 14	13		X	SP
		14	X		SP
15 15C	16C 16	15		X	SP
		16	X		SP
17 17C	18C 18	17		X	SP
		18	X		SP
19 19C	20C 20	19		X	SP
		20	X		

LIMITORQUE
MOV LIMIT SW DEVELOPMENT

ROTOR	CONTACT	VALVE POSITION OPEN – CLOSED	
1	1		SP
1	2		SP
1	3		
1	4		
2	5		
2	6		SP
2	7		SP
2	8		SP
3	9		
3	10		
3	11		SP
3	12		SP
4	13		
4	14		SP
4	15		
4	16		
-	17	TSC-CLG TORQUE SW	SP
-	18	TSO-OPG TORQUE SW	

SOLID LINE DENOTES CONTACT CLOSED



SCHEME 1NCBCF08A
SEE NOTE 5

SCHEME 1BBB38
1HV5132
S.U.NO. AL

NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DWGS. LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER, FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH "Δ" IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP & PERIODIC TESTING.
5. CIRCUITS AFTER THE ISOLATION DEVICES ARE CONSIDERED NON-CLASS 1E.
6. ROTOR NUMBER 2 SHALL BE USED ONLY FOR TORQUE SWITCH BYPASS SERVICE IN OPEN CIRCUIT.

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	REFERENCE DWGS DESCRIPTION	REV
74	C-H	D26M	D26MR13A	1X4DB161-2	P & I DIAGRAM	—
AX	C-H	D26M	D26MR31A	1X3D-AA-F17A	ONE LINE DIAGRAM	—
				1X5DN121-2	LOGIC DIAGRAM	—
				1X5AC01-251	BECHTEL V/P	—

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
9	REVISED PER REA 02-VAA615	07-15-02	ELC	TSL/DRD	EOG	WFP

NUCLEAR SAFETY RELATED B

Q

AUXILIARY FW PUMP P4002 DISCHARGE TRAIN B VALVE

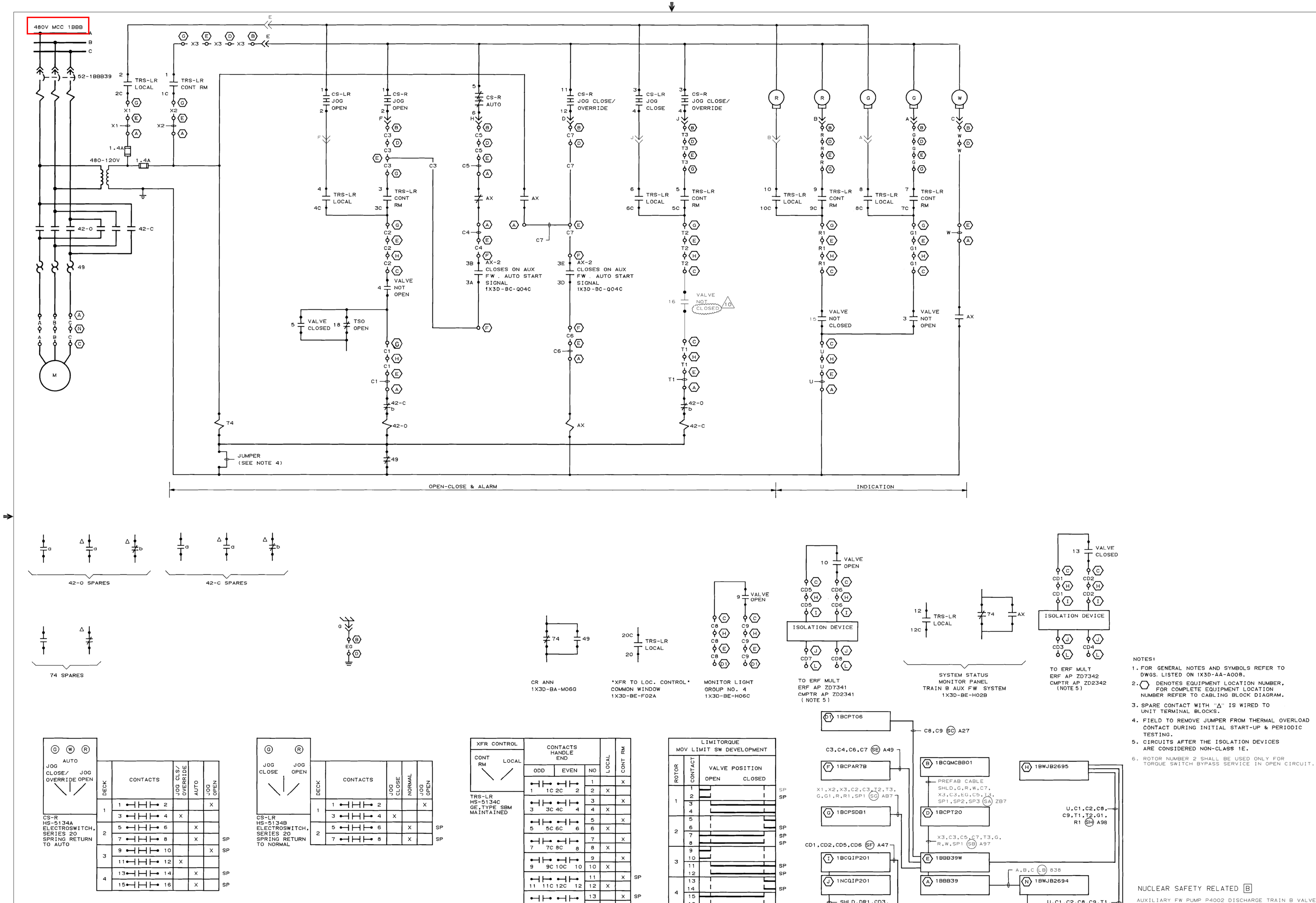
SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
AUX FEEDWATER SYSTEM
1HV-5132

SCALE: NONE
JOB NO.10604
DRAWING NO. 1X3D-BC-F08A
REV. 9

SIZE E 34X44



NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DWGS. LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER, FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH "Δ" IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP & PERIODIC TESTING.
5. CIRCUITS AFTER THE ISOLATION DEVICES ARE CONSIDERED NON-CLASS 1E.
6. ROTOR NUMBER 2 SHALL BE USED ONLY FOR TORQUE SWITCH BYPASS SERVICE IN OPEN CIRCUIT.

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS NUMBER	DESCRIPTION	REV
74	CH	D26M	D26MR13A	1X4DB161-2	P & I DIAGRAM	——
AX	CH	D26M	D26MR31A	1X3D-AA-F17A	ONE LINE DIAGRAM	——
				1X5DN121-2	LOGIC DIAGRAM	——
				1X5AC01-251	BECHTEL V/P	——

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
10	INCORP. PER ABN-38005	11/06/02	ELC	EOG	MWD	X

NUCLEAR SAFETY RELATED B

AUXILIARY FW PUMP P4002 DISCHARGE TRAIN B VALVE Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
AUX FEEDWATER SYSTEM
1HV-5134

SCALE: NONE	DRAWING NO.	REV.
JOB NO.10604	1X3D-BC-F08B	10

SIZE E 34X44

1. 062K3.02 001/LOIT AND LOCT/RO/C/A 4.1/4.4/063K3.02/LO-TA-60012///063K3.01

Initial conditions:

- Unit 1 at 100% reactor power.
- 1AY2A is de-energized.

Current condition:

- Safety Injection is actuated.

Which one of the following completes the following statement?

DG1A __(1)__ automatically start,

and

the Train 'A' sequencer __(2)__ run the Safety Injection load sequence.

	__(1)__	__(2)__
A.	will	will
B.✓	will	will NOT
C.	will NOT	will
D.	will NOT	will NOT

K/A

062 AC Electrical Distribution

K3.02 Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following:

- ED/G

K/A MATCH ANALYSIS

The question tests the candidates knowledge of the effect that a loss of 120VAC Vital AC bus will have on the operation of the ED/G and its associated sequencer during a Safety Injection actuation.

EXPLANATION OF REQUIRED KNOWLEDGE

During a loss of 1AY2A, the train 'A' sequencer is de-energized. Therefore, when a Safety Injection actuation occurs, the SI sequence will not run and no equipment will change state. During an SI actuation, the ED/G receives a start signal from the

sequencer. With 1AY2A de-energized, this start signal will not be generated. However, the ED/G also receives a start signal from SSPS during an SI acutation. This signal will be received since 1AY1A remains energized. The DG will start and run unloaded with its output breaker open.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The ED/G will receives a start signal from SSPS during an SI actuation, independent of the sequencer.

The second part is incorrect. During a loss of 1AY2A, the train 'A' sequencer is de-energized. Therefore, when a Safety Injection actuation occurs, the SI sequence will not run and no equipment will change state. However, candidates frequently confuse the power supply to SSPS slave relays (AY1A) and the power supply to the sequencer (AY2A). If the candidate confuses the power supplies, they may conclude that the sequencer will run the SI sequence.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. During a loss of 1AY2A, the train 'A' sequencer is de-energized. Therefore, when a Safety Injection actuation occurs, the SI sequence will not run and no equipment will change state.

C. Incorrect. Plausible. The first part is incorrect. The ED/G receives a start signal from SSPS during an SI actuation, independent of the sequencer. However, candidates frequently confuse the power supply to SSPS slave relays (AY1A) and the power supply to the sequencer (AY2A). If a candidate reversed the two and believed the ED/G only received a start signal from SSPS during an SI, then it would be reasonable for the candidate to assume the ED/G would not start, but the sequencer would run the SI sequence.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	063K3.02
Importance Rating:	4.1 / 4.4

Technical Reference: ELEMENTARY 1X3D-BH-G03C, Rev 7.0
ONE-LINE 1X3D-AA-G02A, Rev 28.0
ONE-LINE 1X3D-AA-G02C, Rev 14.0

References provided: None

Learning Objective: LO-LP-60324-04 State the effect on SSPS and Sequencer operation on loss of the following:
a. 1AY1A
b. 1AY2A

LO-TA-60012 Respond to a Loss of Vital Instrument Power per 18032-1/2

Question origin: MODIFIED - HL17 NRC # 063K3.01

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7

Comments:

You have completed the test!

1. 063K3.01 001/2/1/LOSS DC DG STARTING/F 3.7/4.1/NEW/H-17 NRC/RO/SRO/EMT/GCW

Unit 1 is at 100% power.

Original Question

- A loss of 125V DC distribution panel 1AD11 occurs.

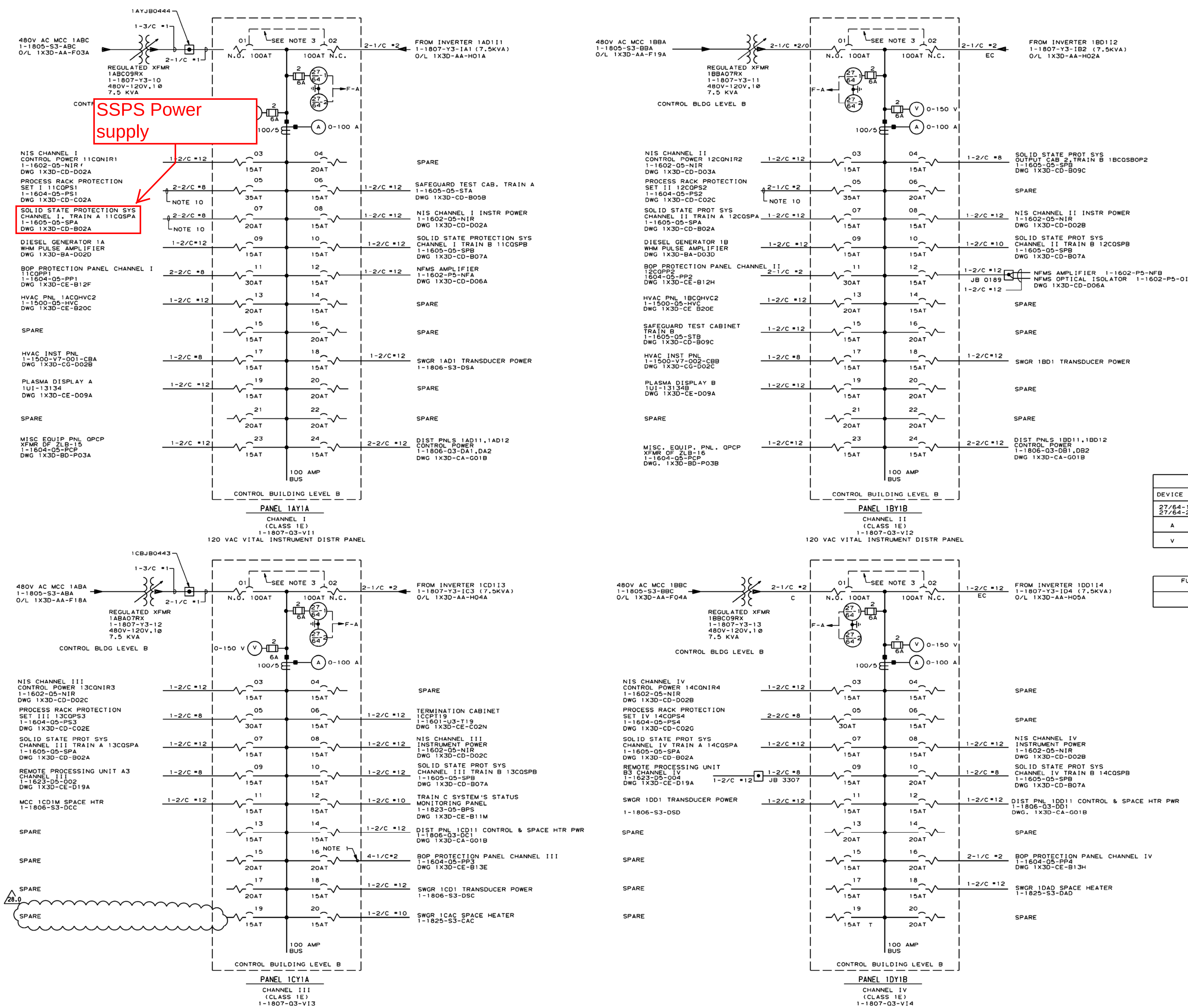
Which one of the following correctly completes the following statement?

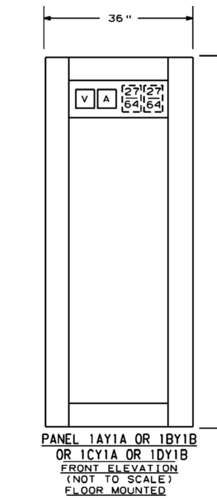
DG1A __(1)__ be started,

and

if DG1A was running prior to the loss of 1AD11, the stop pushbuttons on the QEAB __(2)__ stop the DG.

	__(1)__	__(2)__
A.	can	will NOT
B.✓	can	will
C.	can NOT	will NOT
D.	can NOT	will





LEGEND

DEVICE	MFR	TYPE	DESCRIPTION	REMARKS
27/64-1 27/64-2	ITE	27H	AC GND/UNDV RELAY CAT NO. 211B0265	———
A	GE	AB-40	120V AC PANEL AMMETER	OR EQUAL
V	GE	AB-40	120V AC PANEL VOLTMETER	OR EQUAL

FUNCTION F	DESCRIPTION
A	ANNUNCIATOR

NOTES:

1. ALL BREAKERS SHOWN WITH #2 AWG OR LARGER FIELD RUN CABLE SHALL BE PROVIDED WITH A 6" PIGTAIL OF #2 AWG PER POLE. (#4 AWG FOR BRANCH BREAKERS).
2. BREAKER INTERRUPTING RATINGS ARE 14,000 AMPS AT 120 VAC. (25,000 FOR INPUT BREAKERS).
3. ONLY ONE OF TWO INTERLOCKED BREAKERS FOR EACH PANEL CAN BE CLOSED AT ONE TIME.
4. THE SYSTEM IS UNGROUNDED.
5. ALL BREAKERS, EXCEPT AS NOTED, ARE 15 AT.
6. ALL BREAKERS ARE 2 POLE, 100 AMP FRAME.
7. EACH BREAKER SHALL HAVE AN AUXILIARY "b" BREAKER ALARM CONTACT WHICH SHALL BE WIRED TO AN INTERPOSING RELAY.THE OUTPUT OF THIS RELAY SHALL BE WIRED TO STATES CO. TYPE ZWM TERMINAL BLOCKS INTERNAL IN THE PANEL.
8. UNLESS OTHERWISE NOTED, ALL CABLES ARE ROUTED IN TRAY.
9. ALL #4 AWG & SMALLER CABLE SHALL BE TERMINATED AT THE DISTR. PNLS. USING LUGS MK NOS. PRM,PRQ, OR PRX.
10. CONNECT IN PARALLEL TO ONE 6" PIGTAIL OF #10 AWG.
11. FOR GENERAL NOTES AND SYMBOLS REFER TO DWG. 1X3D-AA-A00A.
12. INDIVIDUAL ELECTRICAL LOADS FOR PANELS 1AY1A, 1BY1B, 1CY1A & 1DY1B ARE LISTED AND TOTALED IN CALCULATION X3CH01.

NUCLEAR SAFETY RELATED A B C D

NO.	VERSIONS	DATE	DR	CHK	APPV
28.0	REVISED PER ABN 1062305601E024, VER.1.0	05/21/12	TSL	RBH	JMR

Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

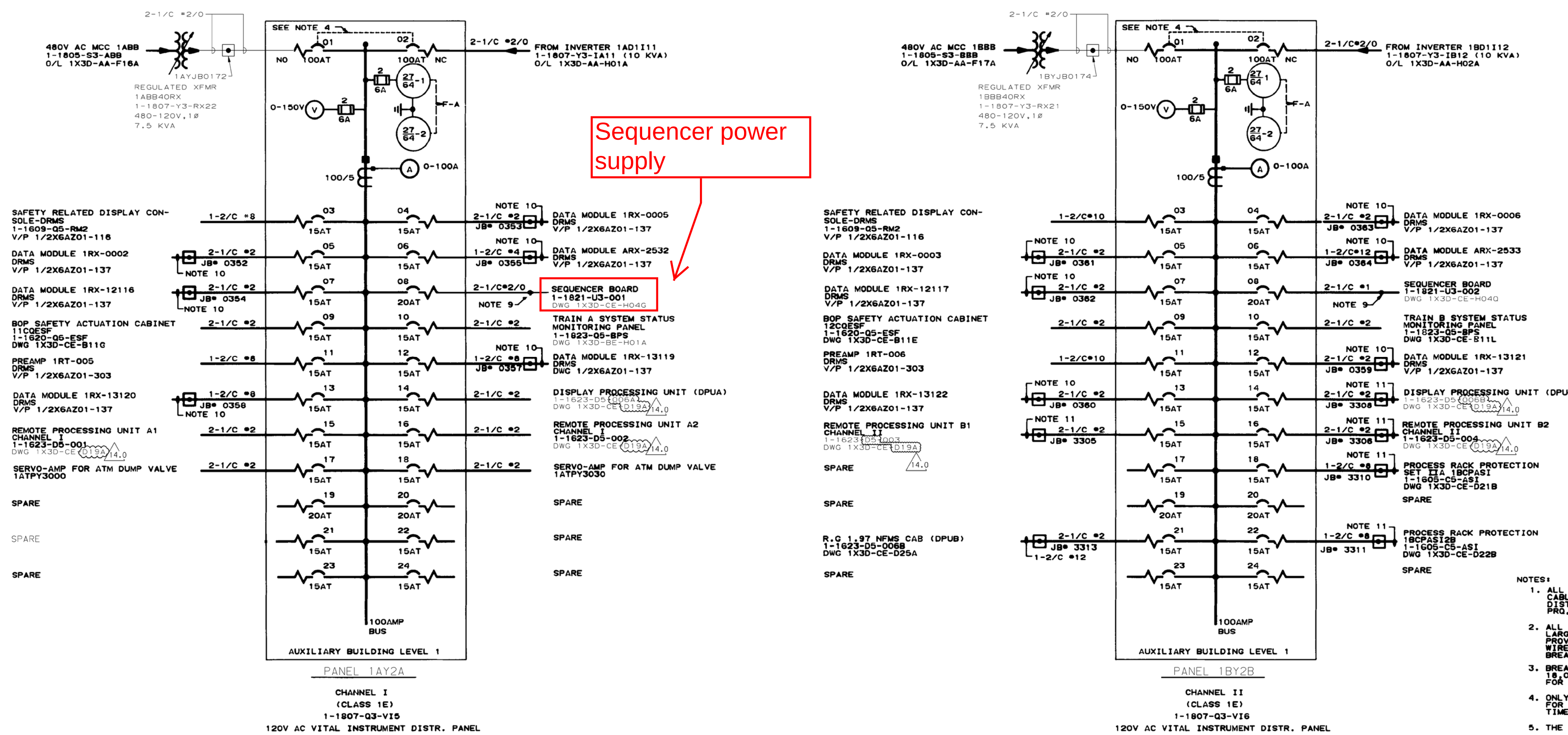
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

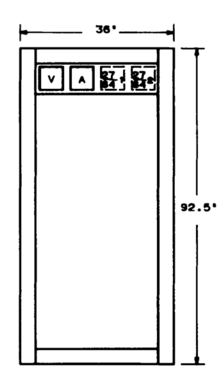
ONE LINE DIAGRAM
120V AC CLASS 1E
VITAL INSTRUMENT DIST PNLS
1AY1A, 1BY1B, 1CY1A & 1DY1B

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-G02A	28.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL





PANEL 1AY2A & 1BY2B
FRONT ELEVATION
(NOT TO SCALE)
FLOOR MOUNTED

LEGEND				
DEVICE	MFR	TYPE	DESCRIPTION	REMARKS
27/64-1 27/64-2	1TE	27H	AC GND/UNDV RELAY CAT# 211B0265	——
V	GE	AB-40	120V AC PNL VOLTMETER	OR EQUAL
A	GE	AB-40	120V AC PNL AMMETER	OR EQUAL

FUNCTION F	DESCRIPTION
A	ANNUNCIATOR

NOTES:

1. ALL #4 AWG AND SMALLER FIELD RUN CABLE SHALL BE TERMINATED AT THE DISTR PNLS USING LUGS MK NOS. PRM, PRQ, OR PRX, AS APPROPRIATE.
2. ALL BREAKERS SHOWN WITH A #2 AWG OR LARGER FIELD RUN CABLE SHALL BE PROVIDED WITH 6" PIGTAIL OF #2 AWG WIRE PER POLE (#4AWG FOR BRANCH BREAKERS).
3. BREAKER INTERRUPTING RATING ARE 18,000 AMPS AT 120V AC.(25,000 AMPS FOR INPUT BREAKER).
4. ONLY ONE OF TWO INTERLOCKED BREAKERS FOR EACH PANEL CAN BE CLOSED AT ONE TIME.
5. THE SYSTEMS ARE UNGROUNDED.
6. ALL BREAKERS ARE 2 POLE, 100 AMP FRAME.
7. EACH BREAKER SHALL HAVE AN AUXILIARY "b" BREAKER ALARM CONTACT WHICH SHALL BE WIRED TO AN INTERPOSING RELAY. THE OUTPUT OF THIS RELAY SHALL BE WIRED TO STATES CO. TYPE ZWM TERMINAL BLOCKS INTERNAL IN THE PANEL.
8. UNLESS OTHERWISE NOTED. ALL CABLES ARE ROUTED IN TRAY.
9. A BUTT SPLICE IS TO BE MADE TO 1-2/C #10 CABLE AT THE LOAD.
10. CABLE 1-2/C #12 SHIELD (A28S) FROM JUNCTION BOX TO LOAD. JUNCTION BOX TO BE LOCATED WITHIN 10 FT OF LOAD.
11. CABLE 1-2/C #12 (A28S) FROM JUNCTION BOX TO LOAD.
12. FOR GENERAL NOTES AND SYMBOLS REFER TO DWG.1X3D-AA-A00A.
13. INDIVIDUAL ELECTRICAL LOADS FOR PANELS 1AY2A & 1BY2B ARE LISTED AND TOTALED IN CALCULATION X3CH01.

NUCLEAR SAFETY RELATED A & B

Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

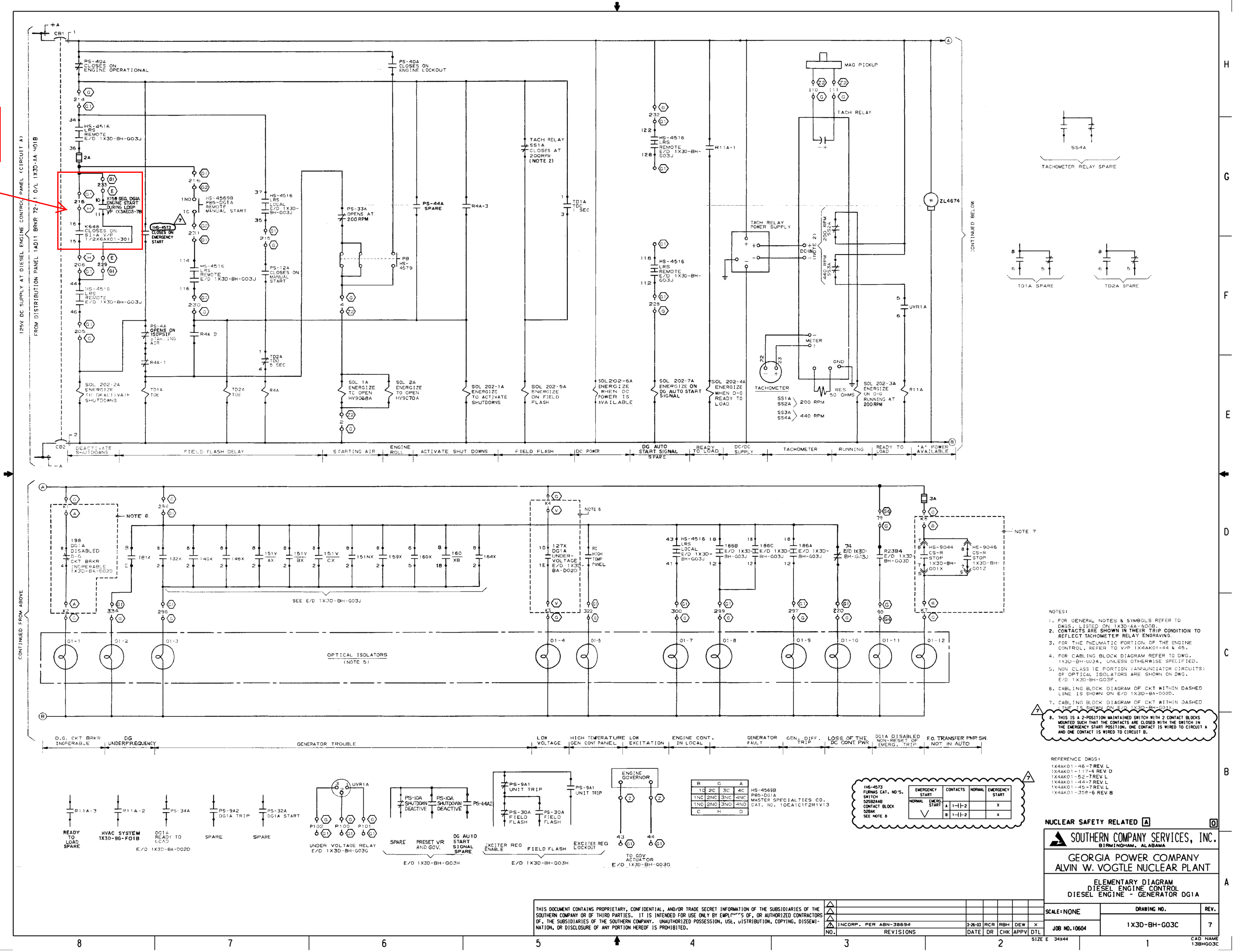
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

120V AC CLASS 1E
VITAL INSTR DISTR PNLS
1-1807-Q3-VI5 & VI6

NO.	VERSIONS	DATE	DR	CHK	APPV
14.0	REVISED PER ABN-81296, VER.1.0	1-25-06	TJS	JEM	ASK

SCALE: NONE
JOB NO.10604
DRAWING NO. 1X3D-AA-G02C
VER. 14.0
SIZE E 34X44
DWG CATEGORY: CRITICAL

ED/G start signals from SSPS and Sequencer



1. 062K4.07 001/LOIT/RO/C/A 2.7/3.1/062K4.07/LO-TA-60012///

Initial conditions:

- Unit 1 is at 100% reactor power.
- DG1A is tagged out for maintenance.

Current condition:

- Unit 1 experiences an LOSP.

Which one of the following completes the following statement?

1CY1A is normally powered by a(an) __(1)__,

and

1CY1A will be __(2)__ after the LOSP occurs.

	__(1)__	__(2)__
A.✔	inverter	energized
B.	inverter	de-energized
C.	regulated transformer	energized
D.	regulated transformer	de-energized

K/A

062 **AC Electrical Distribution**

K4.07 **Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following:**

- One-line diagram of 4 kVAC to 480 VAC distribution, including sources of normal and alternative power.

K/A MATCH ANALYSIS

The question tests the candidates knowledge of AC distribution design features to include how the 125Vdc bus powers 1CY1A via an inverter. Alternately, 1CY1A can be powered from a 480VAC transformer. Both alignments are used in the stem to test the knowledge of the candidate related to the given scenario and the impact on the AC electrical distribution system.

EXPLANATION OF REQUIRED KNOWLEDGE

During a LOSP with the 1A DG tagged out, 1CY1A will energize or de-energize based on the source from which it is powered. Normally, 1CY1A is energized by a DC inverter fed from the 125VDC battery backed bus 1CD1. During the LOSP, 1CD1 will remain energized from its battery bank even thought the two chargers on the bus are de-energized. In this configuration, 1CY1A will remain energized for several hours until battery capacity is depleted. The design basis for battery depletion is 2.75 hours for SI and 4 hours for station blackout.

Alternately, 1CY1A can be energized by a 480VAC regulated transformer. This transformer is 1E powered by a 480V MCC, via a 480V Switchgear, from the 4160V 1E bus 1AA02. During the LOSP, the regulated transformer will de-energize resulting in a loss of power to 1CY1A also. The sequencer will sense the UV condition and perform a load shed. The bus will remain de-energized since the 1A DG is tagged out. Normally, the DG will tie to the bus and the 480V MCC will be loaded back at the first step of the sequence. Sufficient fuel is stored on site to allow the DG to operate for 7 days.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. 1CY1A is normally supplied by an inverter from 1CD1.

The second part is correct. The loss of the 4160V bus will not immediately effect 1CD1 and 1CY1A will remain powered through the inverter from 1CD1.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. The loss of the 4160V bus will not immediately effect 1CD1 and 1CY1A will remain powered through the inverter off 1CD1. However, a candidate with

insufficient knowledge of electrical distribution could remember that the bus is normally aligned to the inverter, but believe the inverter will load shed as part of the LOSP sequence and not re-energize because the DG is tagged out.

C. Incorrect. Plausible. The first part is incorrect. 1CY1A is normally supplied by an inverter from 1CD1. However, a candidate with insufficient knowledge of the distribution system who realizes the bus can be powered from both sources may believe that the regulating transformer is the normal source, and that it would auto transfer to the inverter during a loss of power. Component power sources auto transferring to a battery backup is a common design feature for important systems.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

You have completed the test!

Level:	RO
Tier # / Group #	T2 / G1
K/A#	062K4.07
Importance Rating:	2.7 / 3.1

Technical Reference: SOP 13431-1, Rev 31.1, pages 12 & 15
One-line 1X3D-AA-H04A
One-line 1X3D-AA-G02A

References provided: None

Learning Objective:

LO-PP-01101-05 Describe how a failure of DC control power affects the electrical distribution system and its components.

LO-LP-60324-02 Given that a loss of 120VAC instrument power has occurred to any of the following panels, and given the appropriate plant procedures, describe the operator actions required and why these actions are taken.
e. 1CY1A

LO-LP-60329-02 Given the appropriate drawings, logics, and/or procedures, describe how the plant will respond to a loss of the following 125V DC vital buses:
c. 1CD1

LO-TA-01023 Draw the electrical distribution system

LO-TA-01025 Energizing A Vital Instrument Distribution Panel

LO-TA-60012 Respond to a Loss of Vital Instrument Power per 18032-1/2

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7

Comments: Question appears to match the KA.
The last paragraph of "explanation of required knowledge" seems to be missing something between the end of that page and the middle of the sentence that starts on the next page.
It seems implausible that a candidate would not know that 1CY1A cannot simultaneously have alternate and normal supply breakers closed - are there many examples of 120VAC distribution panels at Vogtle where this is the case? If not, there are two implausible distractors.
The first question asked appears to be somewhat backwards logic - I would recommend stating what the

source is for the power (normal or alternate, rather than inverter vs regulating transformer), and then ask if panel 1CY1A is energized or deenergized. This way, the applicant would have to know which source (inverter or regulating transformer) is the normal and which is the alternate source. Additionally, it gets away from teaching in the question (i.e., the way the question is framed, they know that one of the sources of power keeps the panel energized, and if I had to take a guess not knowing anything about the system, I would have guessed the inverter).
-JAT 12/19/2013 (U/E)

The new question incorporates the above suggestion, however, more information in the stem may be required.
- JAT 2/4/2014

You have completed the test!

INITIALS

4.2 SYSTEM OPERATION

4.2.1 Transferring Vital Instrument Distribution Panel To Alternate Source

CAUTIONS

- A Vital Panel should not be transferred to its alternate source if the battery is not connected to the 125V DC bus. ☐
- 1-HV-15214 will close when 1DY1B is swapped to the alternate source. 1-HV-8160 will close when 1CY1A is swapped to its alternate source. Closure is permissible during solid plant conditions for short duration since PSV-8117 provides pressure relief for low pressure letdown piping. 1-HV-15214 and 1-HV-8160 should be re-opened after their associated bus is re-energized. ☐
- A reactor trip signal will be generated from Source range and Intermediate Range instrumentation if 1AY1A or 1BY1B is transferred below P-10. ☐

4.2.1.1 To preclude any unwanted actuations from opening Instrument Distribution Panel AC Breaker, **review** the appropriate attachment of 18032-1 “Loss of 120 Volt AC Instrument Power” and the 120V AC Load Database for the Vital Instrument Panel to be transferred, to determine what equipment will be impacted and any contingency actions required. ________

4.2.1.2 If transferring 1AY1A to alternate source, perform attachment 1 as directed. ________

4.2.1.3 If transferring 1BY1B to alternate source, perform attachment 2 as directed. ________

4.2.1.4 IF transferring 1DY1B, perform either a or b as directed by SS:

a. IF it is desired to keep CVCS letdown in service, **perform** Attachment 3. ________

b. IF CVCS Letdown will be removed from service, **continue** to step 4.2.1.6. ________

INITIALS

4.2.1.12 IF transferring 1CY1A OR 1DY1B, **remove** SGBD and CVCS Letdown from service to prevent auto isolation from occurring:

- SGBD **removed** from service ________

- CVCS Letdown **removed** from service ________

NOTE

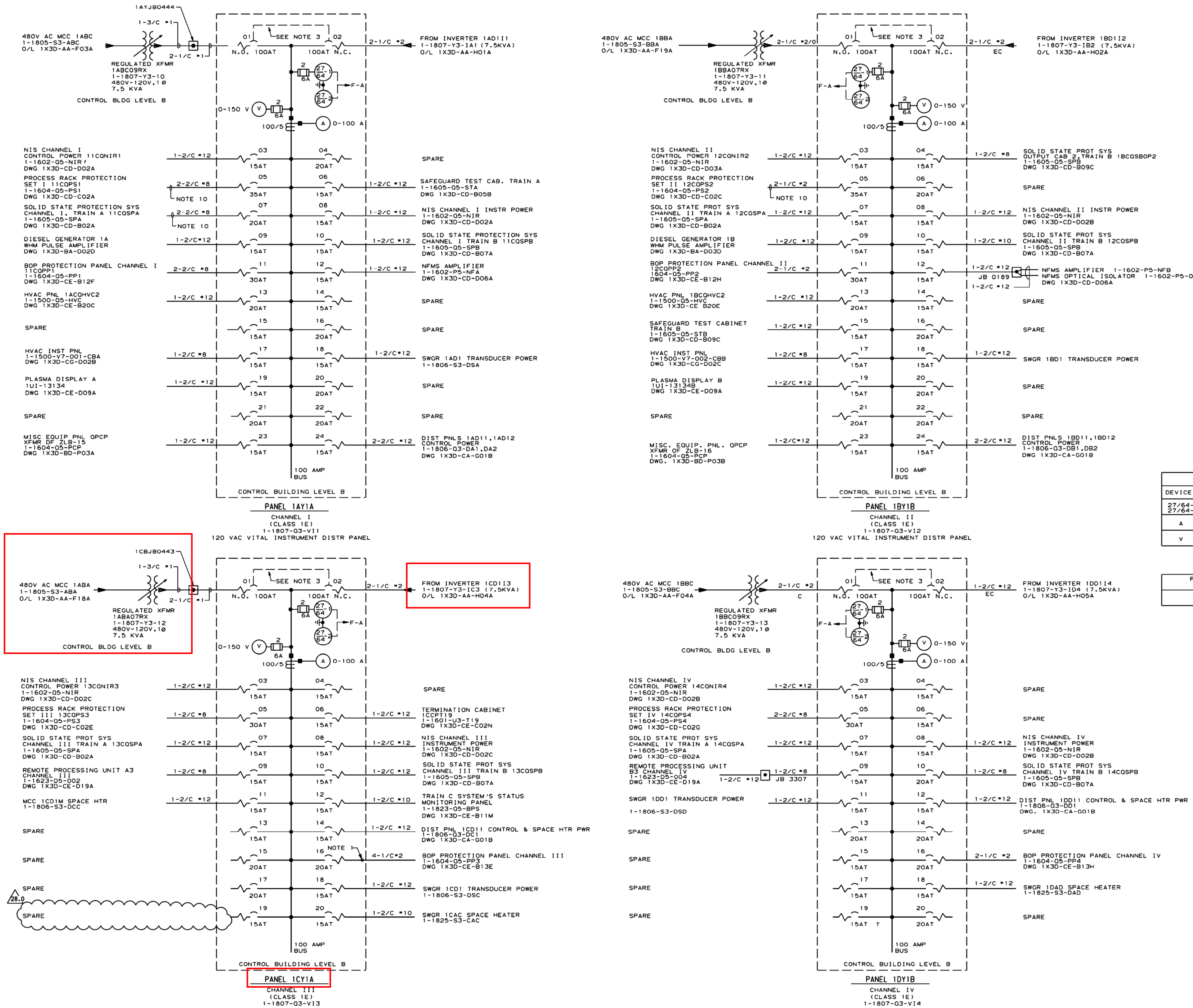
Steps 4.2.1.13 and 4.2.1.14 should be performed as quickly as possible to minimize the time the Instrument Distribution Panel is de-energized. ☐

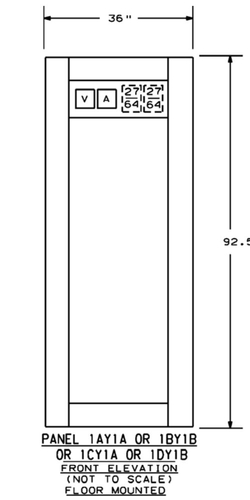
4.2.1.13 **Open** Instrument Distribution Panel AC Breaker: ________

PANEL	PANEL AC BKR
1AY1A	1AY1A-02
1AY2A	1AY2A-02
1BY1B	1BY1B-02
1BY2B	1BY2B-02
1CY1A	1CY1A-02
1DY1B	1DY1B-02

4.2.1.14 **Close** the Instrument Distribution Panel AC Breaker from the regulated source: ________

PANEL	PANEL AC BKR
1AY1A	1AY1A-01
1AY2A	1AY2A-01
1BY1B	1BY1B-01
1BY2B	1BY2B-01
1CY1A	1CY1A-01
1DY1B	1DY1B-01





LEGEND				
DEVICE	MFR	TYPE	DESCRIPTION	REMARKS
27/64-1 27/64-2	ITE	27H	AC GND/UNDV RELAY CAT NO. 211B0265	———
A	GE	AB-40	120V AC PANEL AMMETER	OR EQUAL
V	GE	AB-40	120V AC PANEL VOLTMETER	OR EQUAL

FUNCTION F	DESCRIPTION
A	ANNUNCIATOR

NOTES:

1. ALL BREAKERS SHOWN WITH #2 AWG OR LARGER FIELD RUN CABLE SHALL BE PROVIDED WITH A 6" PIGTAIL OF #2 AWG PER POLE. (#4 AWG FOR BRANCH BREAKERS).
2. BREAKER INTERRUPTING RATINGS ARE 14,000 AMPS AT 120 VAC. (25,000 FOR INPUT BREAKERS).
3. ONLY ONE OF TWO INTERLOCKED BREAKERS FOR EACH PANEL CAN BE CLOSED AT ONE TIME.
4. THE SYSTEM IS UNGROUNDED.
5. ALL BREAKERS, EXCEPT AS NOTED, ARE 15 AT.
6. ALL BREAKERS ARE 2 POLE, 100 AMP FRAME.
7. EACH BREAKER SHALL HAVE AN AUXILIARY "b" BREAKER ALARM CONTACT WHICH SHALL BE WIRED TO AN INTERPOSING RELAY.THE OUTPUT OF THIS RELAY SHALL BE WIRED TO STATES CO. TYPE ZWM TERMINAL BLOCKS INTERNAL IN THE PANEL.
8. UNLESS OTHERWISE NOTED, ALL CABLES ARE ROUTED IN TRAY.
9. ALL #4 AWG & SMALLER CABLE SHALL BE TERMINATED AT THE DISTR. PNLS. USING LUGS MK NOS. PRM,PRQ, OR PRX.
10. CONNECT IN PARALLEL TO ONE 6" PIGTAIL OF #10 AWG.
11. FOR GENERAL NOTES AND SYMBOLS REFER TO DWG. 1X3D-AA-A00A.
12. INDIVIDUAL ELECTRICAL LOADS FOR PANELS 1AY1A, 1BY1B, 1CY1A & 1DY1B ARE LISTED AND TOTALED IN CALCULATION X3CH01.

NUCLEAR SAFETY RELATED A B C D

NO.	VERSIONS	DATE	DR	CHK	APPV
28.0	REVISED PER ABN 1062305601E024, VER.1.0	05/21/12	TSL	RBH	JMR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

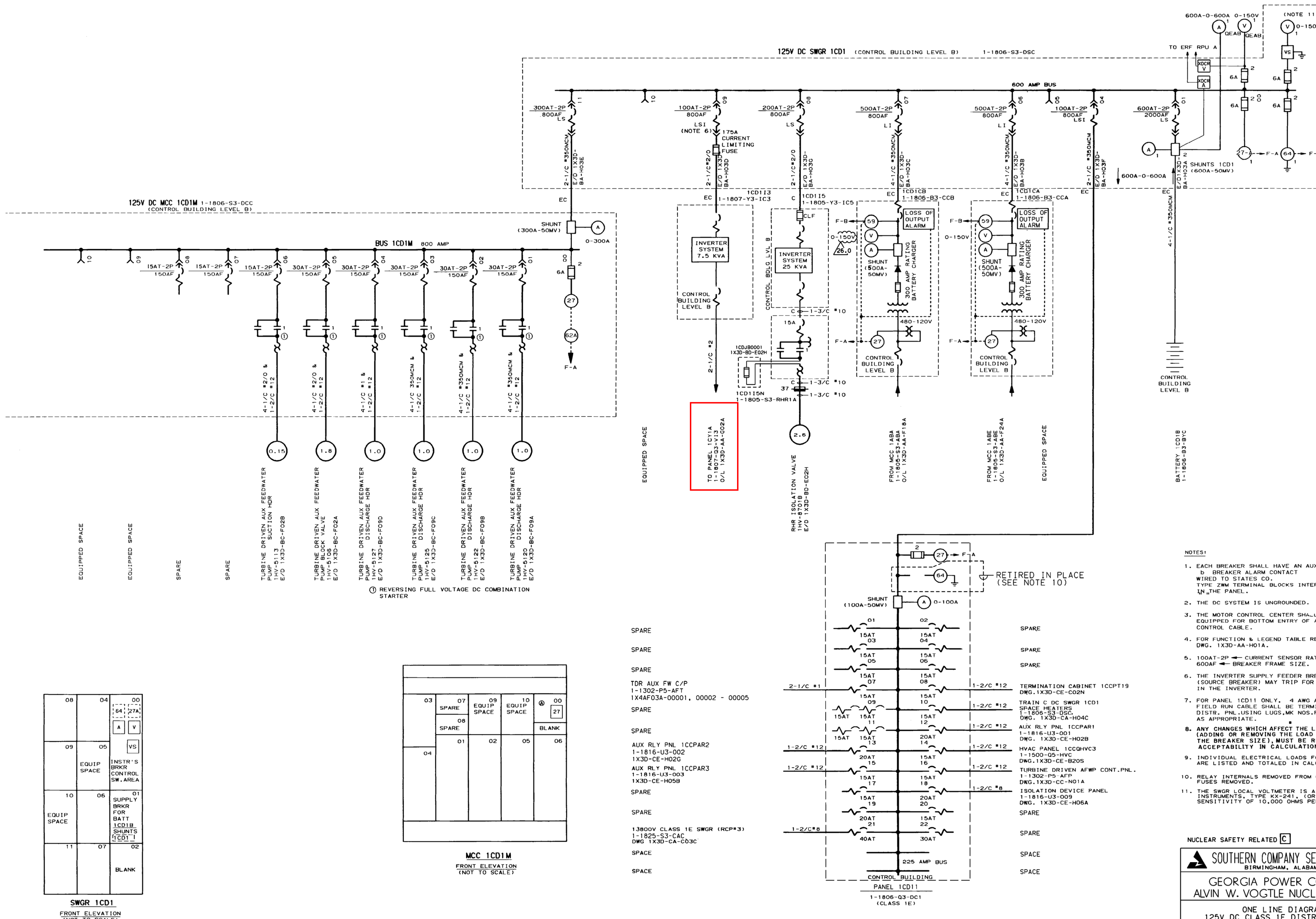
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ONE LINE DIAGRAM
120V AC CLASS 1E
VITAL INSTRUMENT DIST PNLS
1AY1A, 1BY1B, 1CY1A & 1DY1B

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-G02A	28.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. EACH BREAKER SHALL HAVE AN AUXILIARY b BREAKER ALARM CONTACT WIRED TO STATES CO. TYPE ZWM TERMINAL BLOCKS INTERNAL IN THE PANEL.
2. THE DC SYSTEM IS UNGROUNDED.
3. THE MOTOR CONTROL CENTER SHALL BE EQUIPPED FOR BOTTOM ENTRY OF ALL POWER AND CONTROL CABLE.
4. FOR FUNCTION & LEGEND TABLE REFER TO DWG. 1X3D-AA-H01A.
5. 100AT-2P ← CURRENT SENSOR RATING 600AF ← BREAKER FRAME SIZE.
6. THE INVERTER SUPPLY FEEDER BREAKER (SOURCE BREAKER) MAY TRIP FOR A FAULT IN THE INVERTER.
7. FOR PANEL 1CD11 ONLY, 4 AWG AND SMALLER FIELD RUN CABLE SHALL BE TERMINATED AT THE DISTR. PNL.USING LUGS,MK NOS.PRM,PRQ,OR PRX AS APPROPRIATE.
8. ANY CHANGES WHICH AFFECT THE LARGEST LOAD (ADDING OR REMOVING THE LOAD OR CHANGING THE BREAKER SIZE), MUST BE REVIEWED FOR ACCEPTABILITY IN CALCULATION X3CT08.
9. INDIVIDUAL ELECTRICAL LOADS FOR PANELS 1CD11 ARE LISTED AND TOTALED IN CALCULATION X3CF12.
10. RELAY INTERNALS REMOVED FROM CASE, AND FUSES REMOVED.
11. THE SWGR LOCAL VOLTMETER IS A WESCHLER INSTRUMENTS, TYPE KX-241, (OR EQUAL) WITH A SENSITIVITY OF 10,000 OHMS PER VOLT.

NUCLEAR SAFETY RELATED C

NO.	VERSIONS	DATE	DR	CHK	APPV
26.0	REVISED PER ABN 1062305601E023, VER.1.0	05/21/15	TSL	RBH	JMR

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ONE LINE DIAGRAM
125V DC CLASS 1E DISTR. TRAIN C
1-1806-S3-DSC, 1-1806-S3-DCC

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X3D-AA-H04A	26.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. 063A1.01 001/LOIT/RO/C/A 2.5/3.3/063A1.01/LO-TA-37018///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- Unit 1 reactor is tripped.
- 19100-C, "Loss of All AC Power," is in progress.

Which one of the following completes the following statement?

Based on the given conditions, 1CD1B batteries have sufficient capacity to supply the required loads for __(1)__ hours,

and

in order to minimize drain on the batteries, control of the TDAFW feed flow to the SGs is done using __(2)__.

	__(1)__	__(2)__
A.✔	4	1-PDIC-5180A, TDAFW pump speed controller
B.	4	TDAFW pump discharge throttle valves
C.	6.75	1-PDIC-5180A, TDAFW pump speed controller
D.	6.75	TDAFW pump discharge throttle valves

K/A

063 DC Electrical Distribution

A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including:

- Battery capacity as it is affected by discharge rate.

K/A MATCH ANALYSIS

The question addresses how battery load and operator action determine discharge rates. This includes the procedural actions associated with TDAFW valve operation that were incorporated to extend the life on the batteries.

EXPLANATION OF REQUIRED KNOWLEDGE

Per TS 3.8.4 BASES, the 1E batteries are sized to have sufficient capacity to supply the

required loads for 2.75 hours during a LOSP with a LOCA and for 4.0 hrs during a station blackout. In order to meet this requirement, non-required loads are stripped from the bus and operation of DC components is minimized. Per SOP 13610-1 Limitation 2.1.8, if it is necessary to minimize drain on the C-Train batteries during emergency conditions, TDAFW pump speed may be adjusted in lieu of, or in addition to, using the pump discharge throttle valves to control AFW flow to the SGs.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The 1E batteries are sized to have sufficient capacity to supply the required loads for 4.0 hrs during a station blackout.

The second part is correct. If it is necessary to minimize drain on the C-Train batteries during emergency conditions, AFW flow to the SGs is controlled by adjusting TDAFW pump speed in lieu of using the pump discharge throttle valves.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is incorrect. If it is necessary to minimize drain on the C-Train batteries during emergency conditions, AFW flow to the SGs is controlled by adjusting TDAFW pump speed in lieu of using the pump discharge throttle valves. However, the throttling of discharge valves is the normal means for adjusting flow. EOP 19100-C steps 21 and 24 even discuss closing the throttle valves "one at a time". Therefore, it is reasonable that a candidate with insufficient knowledge of the best method to conserve battery power would be reluctant to reduce TDAFW speed and would in turn adjust each throttle valve.

C. Incorrect. Plausible. The first part is incorrect. The 1E batteries are sized to have sufficient capacity to supply the required loads for 2.75 hours during a LOSP with a LOCA and for 4.0 hrs during a station blackout. However, a candidate with insufficient knowledge of battery capacities could combine the 2.75 hr capacity for LOSP with a LOCA, and the 4 hr station blackout capacity, not realizing the 2.75 hr capacity accounted for both events.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	063A1.01
Importance Rating:	2.5 / 3.3

Technical Reference: EOP 19100-C, Rev 38.1, pages 15 & 17
SOP 13610-1, Rev 50.4, page 9
TS 3.8.4 Bases, Rev 2-3/05, page B 3.8.4-2

References provided: None

Learning Objective:

LO-LP-41201-01 Briefly describe the principle of operation of a lead acid wet cell. Include both charging and discharging.

LO-LP-39212-02 Given a set of Tech Specs and the bases, determine for a specific set of plant conditions, equipment availability, and operational mode:
a. Whether any Tech Spec LCOs of section 3.8 are exceeded.
b. The required actions for all section 3.8 LCOs.

LO-LP-37031-08 Using EOP 19100-C as a guide, briefly describe how each step is accomplished.

LO-PP-20101-09 Determine the impact to AFW system operation and the overall integrated plant operations to the following types of power supply failures:
a. U/V condition on either AA02 or BA03 with the bus being re-energized from the EDG while at 100% power.
b. U/V condition on either AA02 or BA03 with the bus remaining de-energized while at 100% power.
c. Loss of a 120 VAC 1E vital instrument bus.
d. Loss of a 125 VDC 1E bus
e. Loss of All AC Power

LO-TA-37018 Respond to a Loss of All AC Power per 19100-C

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.8 / 41.10

Comments:

You have completed the test!

BASES

BACKGROUND (continued)

Batteries are sized in accordance with IEEE 485 (Ref. 3) to have sufficient capacity to supply the required loads for a loss of coolant/ loss of offsite power (LOCA/LOSP) duration of 2 3/4 hours and a station blackout (SBO) duration of 4 hours. For LOSP/LOCA, they are sized at a minimum temperature of 70°F; their initial capacity was increased by 10% for load growth and 25% for aging. The required final (end of duty cycle and end of life) battery cell voltages for each load group have been analyzed to demonstrate that adequate voltage is provided to the loads. The battery voltage specifications are discussed in detail for each load group in FSAR, Chapter 8 (Ref. 4).

The battery cells are of flooded lead acid construction with a nominal specific gravity of 1.215. This specific gravity corresponds to an open circuit battery voltage of approximately 121.8 V for a 59 cell battery (i.e., cell voltage of 2.065 volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. Once fully charged with its open circuit voltage ≥ 2.065 Vpc, the battery cell will maintain its capacity for 30 days without further charging per manufacturer's instructions. Optimal long term performance however, is obtained by maintaining a float voltage 2.20 to 2.25 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of 2.23 Vpc corresponds to a total float voltage output of 131.6 V for a 59 cell battery as discussed in the FSAR, Chapter 8 (Ref. 4).

Each 125 VDC battery is provided with two battery chargers, each of which is sized to supply the continuous (long term) demand on its associated DC system while providing sufficient power to replace 110% of the equivalent ampere-hours removed from the battery during a design basis battery discharge cycle within a 12 hour period after charger input power is restored. Normally, both battery chargers are on line with load sharing circuitry to ensure that the DC load is properly shared between the two chargers. Only one charger is required OPERABLE to support the associated DC power system. The sizing of each battery charger meets the requirements of IEEE 308 (Ref. 1) and Regulatory Guide 1.32 (Ref. 5).

The battery chargers are normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

(continued)

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20. Check for faulted SG(s):	20. Go to Step 23.

ANY SG PRESSURE LOWERING IN AN UNCONTROLLED MANNER.

-OR-

ANY SG COMPLETELY DEPRESSURIZED.

21. Isolate faulted SG(s):

a. Close the TDAFW Throttle Valves on affected SG(s) one at a time:

HV-5122 (SG 1)

HV-5125 (SG 2)

HV-5127 (SG 3)

HV-5120 (SG 4)

b. Close only one TDAFW Pump Steam Supply Valve from affected SG(s):

HV-3009 (SG 1)

-OR-

HV-3019 (SG 2)

c. Verify affected SG ARV(s) closed:

PV-3000 (SG 1)

PV-3010 (SG 2)

PV-3020 (SG 3)

PV-3030 (SG 4)

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

*24. Isolate ruptured SG(s):

a. Close the TDAFW Throttle Valves on affected SG(s) one at a time:

HV-5122 (SG 1)

HV-5125 (SG 2)

HV-5127 (SG 3)

HV-5120 (SG 4)

b. Close only one TDAFW Steam Supply Valve from affected SG(s):

HV-3009 (SG 1)

-OR-

HV-3019 (SG 2)

c. Limit ruptured SG ARV(s) operation:

Adjust controller setpoint to 1160 psig (pot setting 7.73).

WHEN SG pressure is less than 1160 psig,
THEN verify SG ARV(s) closed.

IF SG ARV(s) can NOT be closed,
THEN locally isolate SG ARV(s).

d. Locally close the MDAFW Throttle Valves on affected SG(s) using ATTACHMENT 6.

e. Verify ruptured SG(s) remains isolated during subsequent recovery actions unless needed for RCS cooldown.

INITIALS

2.1.6 Manual reset of the 186 Lockout Relay is required on the MDAFW Pump Motor Feeder Breakers on phase or ground overcurrent. Also, the System Status Monitor Panel will indicate BYPASSED until the lockout relay is reset. ________

2.1.7 TDAFW and MDAFW Pumps are normally aligned to CST-1. If CST-1 is rendered inoperable, CST supply and mini-flow valves should be aligned to CST-2. (SNC12990,1987311237) ________

2.1.8 During emergency conditions if it is necessary to minimize drain on the C-Train Batteries, TDAFW Pump speed may be adjusted in lieu of, or in addition to, using the pump discharge throttle valves to control AFW flow to the SGs. In this case, TDAFW Pump discharge pressure should be adjusted to ensure the required feed flow is supplied to the SG with the highest pressure. ________

2.1.9 Thirty (30) seconds after TDAFW Pump Steam Supply 1-HV-5106 begins to open, turbine speed less than 175 rpm will cause the governor controller to go into TRIP MODE. This will cause ALB16-E03 AFW TURB TROUBLE to illuminate, and the controller will not provide startup functions. 1-HV-5106 must be closed and then the Trip and Throttle Valve must be cycled open to reset the controller. ________

2.1.10 Handswitches and indicators mentioned in this procedure are located on Panel QMCB unless otherwise stated. ________

2.1.11 It is not desirable to remove the TDAFW pump from service on either unit when work is in progress in the Low Voltage Switchyard or on any Reserve Auxiliary Transformer. ________

2.1.12 MDAFW Pump mini-flow valve operation should be periodically monitored when operating throttle valves. ________

1. 064G2.1.31 001/LOIT AND LOCT/RO/M/F 4.6/4.3/064G2.1.31/LO-TA-11015///

Given the following:

- Unit 1 is at 100% reactor power.
- DG1A is in its normal standby alignment.

Which one of the following completes the following statement?

The DSL GEN 1A UNIT/PARALLEL switch, 1HS-4414B, on the QEAB is in the __(1)__ mode of operation,

and

its associated blue indicating light is __(2)__.

	__(1)__	__(2)__
A.✓	unit	lit
B.	unit	NOT lit
C.	parallel	lit
D.	parallel	NOT lit

K/A

064 Emergency Diesel Generator

G2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

K/A MATCH ANALYSIS

The questions tests the candidate's ability to locate the control room switch and light indication for the Emergency D/G unit selector switch and determine the correct standby position and light indication.

EXPLANATION OF REQUIRED KNOWLEDGE

Per Control Room Rounds 11874-1, the Emergency D/G mode selector switch must be in the UNIT position during standby alignment. Per SOP 13145A-1 step 4.3.1.1, during shutdown of the D/G, the mode selector switch is momentarily placed in the UNIT position and the blue indicator light is verified LIT.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. The Emergency D/G mode selector switch must be in the UNIT position during standby alignment.

The second part is correct. With the mode selector switch momentarily placed in the UNIT position, the blue indicator light is LIT.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. With the mode selector switch momentarily placed in the UNIT position, the blue indicator light is LIT. However, candidates frequently confuse the two DG modes and which one the blue light LIT represents. Additionally, the original Vogtle design had the blue light LIT for parallel mode. It was subsequently changed to LIT in unit mode due to difficulty in verifying standby alignment versus a blown bulb.

C. Incorrect. Plausible. The first part is incorrect. The Emergency D/G mode selector switch must be in the UNIT position during standby alignment. However, candidates frequently confuse the two DG modes. Additionally, an LOSP signal places the DG in unit mode. The candidate may believe that the normal alignment would be parallel, since that is what mode the DG is most frequently operated in during survelliances and that it will shift to unit mode during and LOSP.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	064G2.1.31
Importance Rating:	4.6 / 4.3

Technical Reference: SOP 13145A-1, Rev 6.1, page 32
LINEUP 11874-1, Rev 77.0, page 6

References provided: None

Learning Objective:

LO-TA-01022	Transfer From RAT to SAT per 13418-C
LO-TA-11007	Parallel Normal Incoming Source to 4160V Bus Being Supplied from Diesel Generator using 13427A/B-1/2
LO-TA-11014A	Perform emergency Diesel Generator operability test using 14980A/B-1/2
LO-TA-11015	Prepare Diesel Generator for Automatic Operation using 13145-1/2
LO-PP-11101-49	For the following controls on the Generator Control Panel and/or the QEAB, describe the response of the diesel generator to the selection of each position:

a. Local/Remote switch
b. Speed RAISE/LOWER switch (pushbutton)
c. Exciter Enable pushbutton
d. Emergency Shutdown pushbutton
e. Delete
f. Field Flash pushbutton
g. Voltage Control RAISE/LOWER switch (pushbutton)
h. Unit/Parallel switch
i. Exciter Permissive Switch
j. Load Pot

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.10 / 45.12

Comments:

You have completed the test!

INITIALS

4.3 SHUTDOWN

4.3.1 Stopping Train A Diesel Generator

4.3.1.1 IF stopping DG1A from the QEAB, perform the following:

a. At the Generator Control Panel, **verify** LOCAL-REMOTE Switch 1HS-4516 to REMOTE. ________

b. Momentarily **place** DSL GEN 1A UNIT/PARALLEL switch 1HS-4414B to UNIT and **check** the blue DSL GEN 1A UNIT MODE FAST START light is lit. ________

> NOTE
>
> The DG must idle for 30 seconds after the UNIT/PARALLEL switch is placed in UNIT to verify that the Governor Slow Start timer can time out and thus permit the DG to Fast Start after shutdown. ☐

> ALB35 F06 DG1A SWITCH NOT IN AUTO

c. At MCC 1NBI **place** Lube Oil Circulating Pump handswitch 1HS-24006 in the OFF position. ________

d. WHEN the DG has idled greater than 30 seconds, **depress** STOP pushbutton 1HS-4571B. ________

e. At PDG1, **check** generator voltage drops to zero on FIELD VOLTMETER 1EI-40139. ________

f. At Generator Control Panel PDG1, **check** EXCITER PERMISSIVE switch 1HS-4913 is in the NORMAL position. ________

FIGURE 1
UNIT 1

EMERGENCY SAFEGUARDS EQUIPMENT CHECKLIST (APPLICABILITY - MODES 1 THROUGH 4)

COMPONENT	NUMBER	POSITION	STATUS	SHIFT DAY	SHIFT NIGHT	COMMENT NUMBER
RWST TO CCP A & B SUCTION	1-HS-0112D	AUTO	CLOSED			
RWST TO CCP A & B SUCTION	1-HS-0112E	AUTO	CLOSED			
VCT OUTLET ISOLATION	1-HS-0112B	AUTO	OPEN			
VCT OUTLET ISOLATION	1-HS-0112C	AUTO	OPEN			
BIT DISCH ISOLATION	1-HS-8801A	AUTO	CLOSED			
BIT DISCH ISOLATION	1-HS-8801B	AUTO	CLOSED			
NUCLEAR RECORDER	1-NR-0045 (30)	N/A	NO ALARMS			
TDAFW	1-HS-5106A	AUTO (3)	OPEN / CLOSED			
MDAFW-B	1-HS-5130A	AUTO (3)	ON-OFF			
MDAFW-A	1-HS-5131A	AUTO (3)	ON-OFF			
LP-1 MS SPLY TO AUX FW TD PMP-1	1-HS-3009	N/A	OPEN (3)			
LP-2 MS SPLY TO AUX FW TD PMP-1	1-HS-3019	N/A	OPEN (3)			
SG-1 FROM TDAFW	1-HS-5122A	AUTO	OPEN (3)			
SG-2 FROM TDAFW	1-HS-5125A	AUTO	OPEN (3)			
SG-3 FROM TDAFW	1-HS-5127A	AUTO	OPEN (3)			
SG-4 FROM TDAFW	1-HS-5120A	AUTO	OPEN (3)			
SG-2 FROM MDAFW PMP-B	1-HS-5132A	AUTO	OPEN (3,4)			
SG-3 FROM MDAFW PMP-B	1-HS-5134A	AUTO	OPEN (3,4)			
SG-1 FROM MDAFW PMP-A	1-HS-5139A	AUTO	OPEN (3,4)			
SG-4 FROM MDAFW PMP-A	1-HS-5137A	AUTO	OPEN (3,4)			
AFP DIFF PRESS	1-PDIC-5180A	IN SPEED CONTROL AT 100% (3)	OPERABLE			
DG 1A UNIT/PARALLEL	1-HS-4414B	UNIT	N/A			
DG 1B UNIT/PARALLEL	1-HS-4452B	UNIT	N/A			
QHVC	ALL ZLB's	N/A	POWER AVAILABLE (5)			
RX CAVITY CONCRETE TEMP INDICATOR	1-TJI-12270 (30)	N/A	NO ALARMS			
CTB CLG UNIT FAN-1 LOW SPEED	1-HS-12582A	AUTO	ON-OFF			
CTB CLG UNIT FAN-1 HIGH SPEED	1-HS-12582D	AUTO	ON-OFF			
CTB CLG UNIT FAN-2 LOW SPEED	1-HS-2582A	AUTO	ON-OFF			
CTB CLG UNIT FAN-2 HIGH SPEED	1-HS-2582D	AUTO	ON-OFF			
CTB CLG UNIT FAN-3 LOW SPEED	1-HS-12583A	AUTO	ON-OFF			
CTB CLG UNIT FAN-3 HIGH SPEED	1-HS-12583D	AUTO	ON-OFF			

(3) NOT REQUIRED IN MODE 4
(4) POSITION MAY BE THROTTLED WHEN RUNNING ASSOCIATED AFW PUMP
(5) ZLBs 18, 19, 36, 37, 38, 43 AND 44 ARE NORMALLY DE-ENERGIZED AND THUS ARE EXCLUDED.
(30) IF RECORDER DISPLAYS “NOT ENOUGH FREE SPACE ON MEDIA” MEDIA CARD SHOULD BE CHANGED PER 10001-C SECTION 5.2.

COMMENTS: ______________________

1. 065AA2.08 001/LOIT/RO/C/A 2.9/3.3/065AA2.08/LO-TA-60007///

Initial condition:

- Unit 1 is at 7% reactor power.

Current condition:

- Instrument air line to 1FV-0121, CVCS Charging Flow Control Valve, is severed.

Which one of the following completes the following statement?

With no operator action, seal injection flow to the RCPs will __(1)__,

and

a reactor trip __(2)__ occur on pressurizer level.

	__(1)__	__(2)__
A.	increase	will
B.✓	increase	will NOT
C.	decrease	will
D.	decrease	will NOT

K/A

065 **Loss of Instrument Air**

AA2.08 **Ability to determine and interpret the following as they apply to the Loss of Instrument Air:**

- Failure modes of air-operated equipment.

K/A MATCH ANALYSIS

The question tests the candidate's ability to identify the failure mode of AOV FV-0121 due to a loss of instrument air. The candidate will demonstrate the ability to determine the integrated system response by predicting how seal injection flow will react, and whether a reactor trip will occur due to pressurizer level response.

EXPLANATION OF REQUIRED KNOWLEDGE

Per P&ID 1X4DB116-1, FV-0121 is a fail-open AOV. The question establishes a scenario where the instrument air line to FV-0121 fails, resulting in maximum charging flow. Backpressure valve HV-182 is down stream of FV-121 and is a manually controlled AOV. As total charging flow increases without HV-182 position changing,

more flow will be directed to the RCP seals. Additionally, an increase in flow through HV-182 to the normal charing nozzle will also occur. The net result is a continuous rise in Pressurizer Level. With reactor power < P-7 (2/4 Power range NI's >10%), a reactor trip will not occur when pressurizer level is >92%.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. When instrument air is lost to FV-0121 the valve will fail open. This action will increase the total charging flow and with HV-0182 throttled to a set position it would force more flow into the RCP seal injection line.

The second part is incorrect. With reactor power < P-7 (2 of 4 Power range NI's >10%), a reactor trip will not occur when pressurizer level is >92%. However, a candidate may not recognize, or may have forgotten, the P-7 interlock and assume the reactor trips on pressurizer level >92%.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. With reactor power < P-7 (2 of 4 Power range NI's >10%), a reactor trip will not occur when pressurizer level is >92%.

C. Incorrect. Plausible. The first part is incorrect. When instrument air is lost to FV-0121 the valve will fail open. This action will increase the total charging flow and with HV-0182 throttled to a set position it will force more flow to the RCP seal injection line. However, if the candidate assumes that HV-182 also loses instrument air pressure, then HV-182 would fail open and seal injection flow would lower. The candidate could also reverse the order of FV-0121 and HV-0182 in the flow path and conclude that seal injection flow would lower.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	065AA2.08
Importance Rating:	2.9 / 3.3

Technical Reference: SOP 13503A-1, Rev 7.2, pages 33 & 35
P&ID 1X4DB116-1, Rev 50.0

References provided: None

Learning Objective:

LO-PP-09100-07 Given that a partial or complete loss of instrument air has occurred, determine how CVCS letdown system will respond and describe the steps to control RCS inventory.

LO-PP-09200-06 Given that a partial or complete loss of instrument air has occurred, determine how the CVCS charging system will respond and describe the steps that are required to control RCS inventory and seal injection.

LO-TA-60007 Respond to a Loss of Instrument Air per 18028-C.

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 43.5 / 45.13

Comments:

You have completed the test!

ATTACHMENT C Sheet 1 of 6

PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

PERMISSIVES

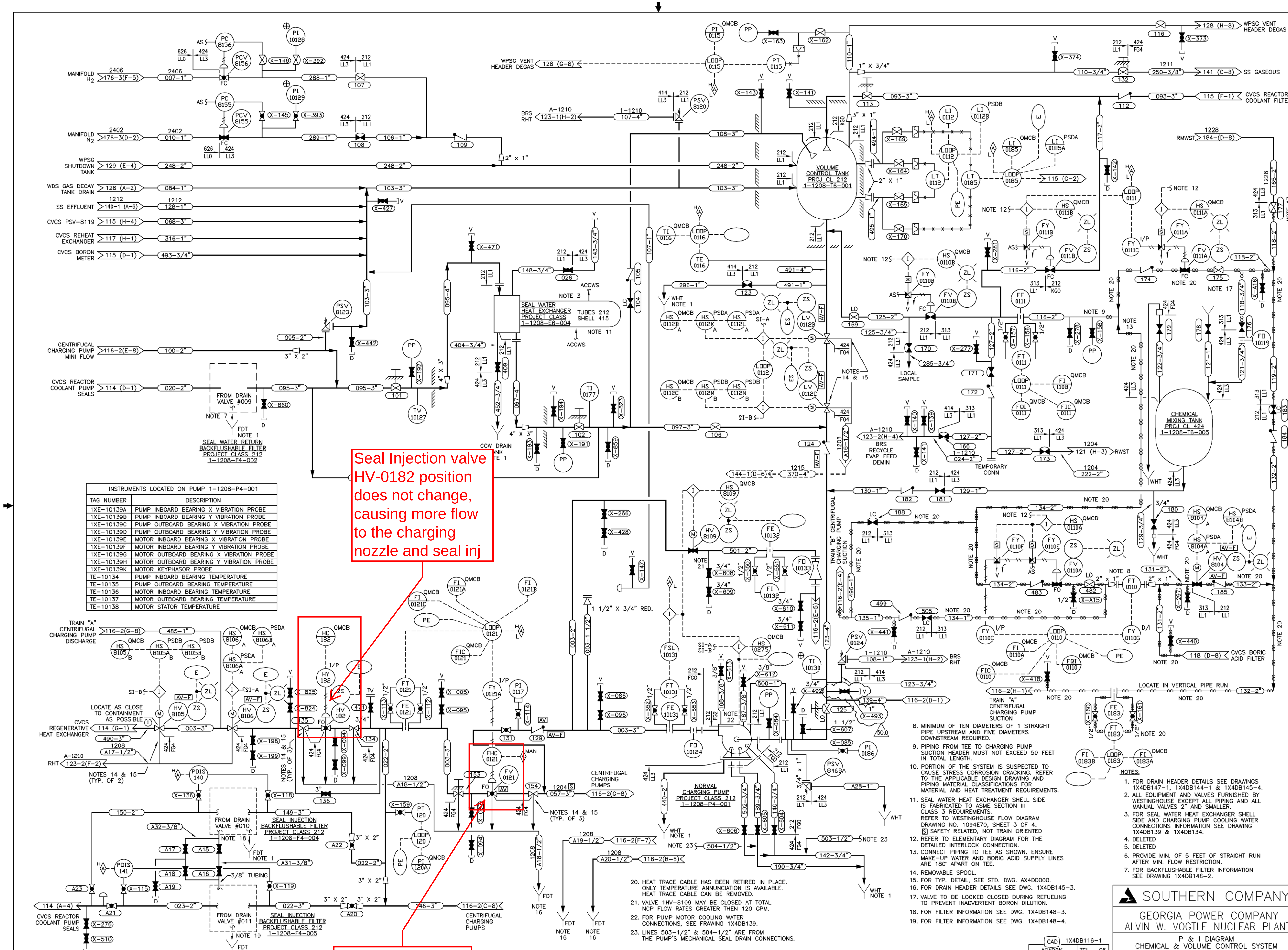
Permissive	Setpoint/Coincidence	Function
P-4	Train related Rx trip & Bypass breaker open	Trips Main Turbine Train A - mechanical Train B - electrical FWI if Lo Tavg (2/4 ≤ 564 °F) present Seals in FWI if caused by SI or P-14 (Hi Hi Level) Must be present to block auto SI after SI reset. P-4 Train A arms Steam Dumps P-4 Train B swaps Steam Dumps to plant trip controller
P-6	1/2 IR Detectors ≥ 2.0 E –5 % Rx Power	Allows manual block of SR High ϕ trip
P-7	P-10 (2/4 PR NIs ≥ 10% Rx power) or P-13 (PT-505 or 506 ≥ 10% turbine power)	Unblocks "At Power" Trips Przr Low Pressure Przr High Level RCS Two Loop Low Flow RCP UF RCP UV
P-8	2/4 PR NIS ≥ 48% Rx power	Enables Single Loop Low Flow Rx Trip
P-9	2/4 PR NIS ≥ 40% Rx power	Enables Turbine trip Rx trip
P-10	2/4 PR NIS ≥ 10% Rx power	Auto block of SR High ϕ trip Enables P-7 Allows manual block of IR rod stop and Hi ϕ trip Allows manual block of PR Hi ϕ trip Lo Setpoint
P-11	2/3 Przr Pressure channels ≤ 2000 psig	Auto enables Lo Przr Press SI & Lo Steamline Press SI/SLI & sends signal to open Accum Isolation Valves when P-11 resets. P-11 allows operator to block PRZR & Steamline low pressure SI & SLI. Also activates "Not Full Open" annunciators for Accumulator MOVs (ALB16; A5, B5, C5 & D5) and HV-8806; (ALB16 E03).
P-12	2/4 NR Tavg ≤ 550 °F	Interlocks Steam Dumps closed (Cooldown Dump Valves PV-507A, B & C) may be reopened by use of Bypass Interlock switches)
P-13	1/2 Turbine Impulse channels ≥ 10%	Enables P-7
P-14	2/4 NR SG Level channels ≥ 82%	Actuates FWI Actuates MFP and Main Turbine trip

ATTACHMENT C Sheet 3 of 6

PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

REACTOR TRIPS

Reactor Trip	Setpoint/Permissive/Blocks	Bases
Manual	1/2 handswitches on QMCB	Ensures capability of operator to manually trip Rx
SR Hi ϕ	1/2 SR Channels ≥ 10^5 cps may be manually blocked above P-6 and auto blocked by P-10	Protection against uncontrolled rod withdrawal from subcritical conditions
IR Hi ϕ	1/2 IR Channels ≥ 25% Power, may be manually blocked above P-10	Protection against uncontrolled rod withdrawal from subcritical conditions
PR Hi ϕ Low Setpoint	2/4 PR Channels ≥ 25% power, may be manually blocked above P-10	Protection against uncontrolled rod withdrawal from subcritical or low power conditions
PR Hi ϕ High Setpoint	2/4 PR Channels ≥ 109% power	Protection against power excursions from all power levels
PR Positive Rate	2/4 PR Channels ≥ +5% in 2 seconds	Protection against rod ejection accident
OTΔT	2/4 ΔT Channels ≥ ΔT Setpoint	Protection against DNB
OPΔT	2/4 ΔT Channels ≥ ΔT Setpoint	Protection against exceeding kW/Ft limitations
Przr Hi Press	2/4 Przr Press Channels ≥ 2385 psig	Ensures RCS integrity (prevents overpressurization)
Przr Lo Press	2/4 Przr Press Channels ≤ 1960 psig (rate compensated) auto blocked below P-7	Protection against DNB
Przr Hi Level	2/3 Przr Level Channels ≥ 92% auto blocked below P-7	Prevents water relief through Przr Code Safety Valves
Single Loop Low Flow	2/3 RCS Flow Channels ≤ 90% on 1/4 loops auto blocked below P-8	DNB protection in the event of loss of flow condition
Two Loop Low Flow	2/3 RCS Flow Channels ≤ 90% on 2/4 loops auto blocked below P-7	DNB protection in the event of loss of flow condition
RCP UV	1/2 13.8 busses ≤ 70% nominal voltage auto blocked below P-7	DNB protection in the event of a loss of forced flow
RCP UF	1/2 13.8 busses ≤ 57.3 Hz auto blocked below P-7, will also trip all RCPs	DNB protection in the event of a loss of forced flow
Turbine Trip / Rx Trip	4/4 Turbine SVs not full open **or** 2/3 ETS pressure switches ≤ 580 psig auto blocked below P-9	Ensures Rx trip upon load reduction outside of design limits (that which can be handled by steam dumps & rod control)
SG Lo Lo Level	2/4 NR SG Level Channels ≤ 38%	Protects against loss of heat sink with sufficient water level to allow for starting delays of AFW
Safety Injection	Any SI, auto or manual	Assures subcriticality during accident conditions
General Warning	GW condition on 2/2 SSPS Trains	Protection against potential loss of auto trip capability



INSTRUMENTS LOCATED ON PUMP 1-1208-P4-001	
TAG NUMBER	DESCRIPTION
1XE-10139A	PUMP INBOARD BEARING X VIBRATION PROBE
1XE-10139B	PUMP INBOARD BEARING Y VIBRATION PROBE
1XE-10139C	PUMP OUTBOARD BEARING X VIBRATION PROBE
1XE-10139D	PUMP OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139E	MOTOR INBOARD BEARING X VIBRATION PROBE
1XE-10139F	MOTOR INBOARD BEARING Y VIBRATION PROBE
1XE-10139G	MOTOR OUTBOARD BEARING X VIBRATION PROBE
1XE-10139H	MOTOR OUTBOARD BEARING Y VIBRATION PROBE
1XE-10139K	MOTOR KEYPHASOR PROBE
TE-10134	PUMP INBOARD BEARING TEMPERATURE
TE-10135	PUMP OUTBOARD BEARING TEMPERATURE
TE-10136	MOTOR INBOARD BEARING TEMPERATURE
TE-10137	MOTOR OUTBOARD BEARING TEMPERATURE
TE-10138	MOTOR STATOR TEMPERATURE

NOTES:

1. FOR DRAIN HEADER DETAILS SEE DRAWINGS 1X4DB147-1, 1X4DB144-1 & 1X4DB145-4.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL MANUAL VALVES 2" AND SMALLER.
3. FOR SEAL WATER HEAT EXCHANGER SHELL SIDE AND CHARGING PUMP COOLING WATER CONNECTIONS INFORMATION SEE DRAWING 1X4DB139 & 1X4DB134.
4. DELETED
5. DELETED
6. PROVIDE MIN. OF 5 FEET OF STRAIGHT RUN AFTER MIN. FLOW RESTRICTION.
7. FOR BACKFLUSHABLE FILTER INFORMATION SEE DRAWING 1X4DB148-2.
8. MINIMUM OF TEN DIAMETERS OF 1 STRAIGHT PIPE UPSTREAM AND FIVE DIAMETERS DOWNSTREAM REQUIRED.
9. PIPING FROM TEE TO CHARGING PUMP SUCTION HEADER MUST NOT EXCEED 50 FEET IN TOTAL LENGTH.
10. PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
11. SEAL WATER HEAT EXCHANGER SHELL SIDE IS FABRICATED TO ASME SECTION III CLASS 3 REQUIREMENTS. REFER TO WESTINGHOUSE FLOW DIAGRAM DRAWING NO. 1094E70, SHEET 3 OF 4.
S SAFETY RELATED, NOT TRAIN ORIENTED
12. REFER TO ELEMENTARY DIAGRAM FOR THE DETAILED INTERLOCK CONNECTION.
13. CONNECT PIPING TO TEE AS SHOWN. ENSURE MAKE-UP WATER AND BORIC ACID SUPPLY LINES ARE 180° APART ON TEE.
14. REMOVABLE SPOOL.
15. FOR TYP. DETAIL, SEE STD. DWG. AX4DD000.
16. FOR DRAIN HEADER DETAILS SEE DWG. 1X4DB145-3.
17. VALVE TO BE LOCKED CLOSED DURING REFUELING TO PREVENT INADVERTENT BORON DILUTION.
18. FOR FILTER INFORMATION SEE DWG. 1X4DB148-3.
19. FOR FILTER INFORMATION SEE DWG. 1X4DB148-4.
20. HEAT TRACE CABLE HAS BEEN RETIRED IN PLACE. ONLY TEMPERATURE ANNUNCIATION IS AVAILABLE. HEAT TRACE CABLE CAN BE REMOVED.
21. VALVE 1HV-8109 MAY BE CLOSED AT TOTAL NCP FLOW RATES GREATER THEN 120 GPM.
22. FOR PUMP MOTOR COOLING WATER CONNECTIONS, SEE FRAWING 1X4DB139.
23. LINES 503-1/2" & 504-1/2" ARE FROM THE PUMP'S MECHANICAL SEAL DRAIN CONNECTIONS.

CAD 1X4DB116-1 ACAD2K TSL - 05

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
50.0	REVISED PER ABN-1081944801M002, VER. 1.0	09/28/12	TSL	CFC	JMR	

SCALE: 0.0 | JOB NO.10604 | DRAWING NO. 1X4DB116-1 | VER. 50.0

SIZE E 34X44 | DRAWING CATEGORY: CRITICAL

1. 069AA2.02 001/LOIT AND LOCT/RO/M/F 3.9/4.4/069AA2.02/LO-LP-39210-01///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- Containment entry is in progress.
- Entry crew reports the inner air lock door seal is damaged.
- The inner air lock door is declared inoperable.

Which one of the following completes the following statement?

To comply with the required action statement of Tech Spec 3.6.2, "Containment Air Locks," __(1)__ air lock door(s) must be verified closed,

and

the Tech Spec 3.6.2 required action must be completed __(2)__.

	__(1)__	__(2)__
A.	both	immediately
B.	both	within 1 hour
C.	only the outer	immediately
D.✓	only the outer	within 1 hour

K/A

069 **Loss of Containment Integrity**

AA2.02 **Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:**

- Verification of automatic and manual means of restoring integrity.

K/A MATCH ANALYSIS

The question tests the candidates ability to determine the status of the containment air lock based on reported damage, and the ability to restore integrity by manual means as directed by Tech Spec 3.6.2.

EXPLANATION OF REQUIRED KNOWLEDGE

Per TS 3.6.2 Cond A, if one air lock door is inoperable, the OPERABLE door must be verified CLOSED within 1 hour and locked CLOSED within 24 hours. The candidate must determine that ONLY the outer door is required to be closed to comply with this RAS.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per TS 3.6.2 Cond A, the outer door must be verified closed within 1 hr. However, the normal alignment would have both doors closed. It is reasonable to assume the candidate may conclude the correct action is to shut both doors even if one is not OPERABLE. The candidate may also consider the situation where the inoperable door is the inner door and in order to perform maintenance on the door the outer must be opened, and that both the doors would need to be shut to maintain integrity.

The second part is incorrect. Per TS 3.6.2 Cond A, the outer door must be verified closed within 1 hr. However, per TS 3.9.4, if the same type of condition were discovered with the plant in lower MODES of operation, the action time would be immediate. Additonally, TS 3.6.2 Cond C for two airlock doors in the same airlock has an action time of immediate.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. One Containment airlock door inoperable per Tech Spec 3.6.2, ‘Containment Air Locks’, Condition ‘A.1’ requires the OPERABLE door be shut within 1 hour.

C. Incorrect. Plausible. The first part is correct. One Containment airlock door inoperable per Tech Spec 3.6.2, ‘Containment Air Locks’, Condition ‘A.1’ requires the OPERABLE door only to be shut, which is the outer door.

The second part is incorrect. See the second part of choice A above.

D. Correct The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T1 / G2
K/A#	069AA2.02
Importance Rating:	3.9 / 4.4

Technical Reference: TS 3.6.2, Amendment No. 96, page 3.6.2-1
TS 3.9.4, Amendment No. 115, page 3.9.4-1

References provided: None

Learning Objective: LO-LP-61209-13 State when containment integrity is required.
LO-LP-39210-01 For any given item in section 3.6 of Tech Specs, be able to:
a. State the LCO.
b. State any one hour or less required actions.

Question origin: BANK

Cognitive Level: M/F

10 CFR Part 55 Content: 43.9 / 45.13

Comments:

You have completed the test!

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Entry and exit are permissible to perform repairs on the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	-----------------NOTES-------------------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit are permissible for 7 days under administrative controls if both air locks are inoperable. --	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Verify the OPERABLE door is closed in the affected air lock.	1 hour
	AND		
	A.2	Lock the OPERABLE door closed in the affected air lock.	24 hours
	AND		
	A.3	------------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----------------------------------	
		Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

1. 071A3.03 001/LOIT AND LOCT/RO/M/F 3.6/3.8/071A3.03/LO-TA-32007//HL-15 AUDIT/

Initial condition:

- Unit 1 Waste Gas Decay Tank release is in progress.

Current condition:

- ARE-0014, Waste Gas Processing Effluent Monitor, is in **Intermediate** alarm.

Which one of the following completes the following statement?

ARV-0014, Plant Vent Radwaste Gas, isolation valve is __(1)__,

and

the position of ARV-0014 is to be verified __(2)__.

A. (1) open

(2) on the IPC in the Control Room

B.✓ (1) open

(2) locally on the Gaseous Waste Panel

C. (1) closed

(2) on the IPC in the Control Room

D. (1) closed

(2) locally on the Gaseous Waste Panel

K/A

071 **Waste Gas Disposal System (WGDS):**

A3.03 **Ability to monitor automatic operation of the Waste Gas Disposal System, including:**

- Radiation monitoring system alarm and actuating signals.

K/A MATCH ANALYSIS

The question tests the candidates ability to monitor the Waste Gas Disposal System release path by determining if an actuation signal has been generated based on the effluent rad monitor system alarm.

EXPLANATION OF REQUIRED KNOWLEDGE

Per SOP 13202-1 step 4.2.2 and ARP 17216-1, a HIGH radiation alarm on ARE-014 produces an actuation signal, which closes ARV-0014 and terminates the release. Since ARE-0014 is in INTERMEDIATE alarm, ARV-0014 should remain open. ARV-0014 position is not available in the main control room. ARV-0014 is monitored and controlled only from the local Waste Gas Panel on 'D' Level of the Aux Bldg. (Ref P&ID 1X4DB129)

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. ARV-0014 will remain open following the intermediate radiation alarm,but will automatically isolate on a high alarm condition.

The second part is incorrect. ARV-0014 is monitored and controlled only from the local Waste Gas Panel on 'D' Level of the Aux Bldg. However, many important valves and some rad monitor flow transmitters can be monitored from the IPC. ARV-0014 and AFT-0014 are not among these.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. ARV-0014 is monitored and controlled only from the local Waste Gas Panel on 'D' Level of the Aux Bldg.

C. Incorrect. Plausible. The first part is incorrect. ARV-0014 will remain open following the intermediate radiation alarm, but will automatically isolate on a high alarm condition. However, the candidate may believe that any alarm, intermediate or high, would result in an isolation signal. Additionally, some procedures associated with releases will direct the operator to isolate the flowpath if an intermediate alarm is recieved. Containment purge is an example of this - reference SOP 13125-1 precaution 2.1.7.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	071A3.03
Importance Rating:	3.6 / 3.8

Technical Reference: ARP 17216-1, Rev 2.3, pages 18 & 19
SOP 13125-1, Rev 54.0, pages 4
SOP 13202-1, Rev 20.1, pages 8-14
P&ID 1X4DB129, Rev 43.0

References provided: None

Learning Objective: LO-PP-32101-09 Describe those automatic actions that occur for each of the following non-safety monitors when its high alarm setpoint is exceeded:
b. Waste Gas Processing System Effluent (ARE-0014)
LO-TA-32007 Verify Proper Automatic actions to high radiation alarms

Question origin: BANK

Cognitive Level: M/F

10 CFR Part 55 Content: 41.7 / 41.13 / 45.5

Comments:

You have completed the test!

NOTE

A containment release permit is considered a batch release. Batch pre-release permits are calculated based upon the radionuclide mix in the sample, the estimated volume of the release and the maximum release rate of the release. These pre-release bounding parameters ensure compliance with the ODCM. Releases therefore must remain within these bounding parameters for the pre-release calculations to be valid. Stopping a containment release before the 'Release may not continue beyond (Date/Time)' indicated on 36022-C Data Sheet 1 ensures the volume to be released does not exceed the pre-release permit volume. ☐

2.1.3 If a Purge and/or Vent is started within the allotted time limit, the release/vent may continue until the 'Release may not continue beyond (Date/Time)' indicated on 36022-C Data Sheet 1. ________

2.1.4 Containment purges may be stopped and subsequently restarted without any sampling and analysis being performed and without Effluent Permit closure following review and approval of chemistry department personnel. Releases may not continue beyond the 'Release may not continue beyond (Date/Time)' indicated on 36022-C Data Sheet 1. ________

2.1.5 To ensure accurate sampling by 1RE-2565, prior to placing CTB Mini-Purge in service, sampling flow valves 1-1609-U4-052 should be verified open and 1-1609-U4-054 closed. ________

2.1.6 To ensure accurate sampling by 1RE-2565, prior to placing CTB Main Purge in service, sampling flow valves 1-1609-U4-054 should be verified open and 1-1609-U4-052 closed. ________

2.1.7 If a valid Alert is received on 1RE-2565C while a Containment Purge or Pressure Relief is ongoing, the release/vent must be terminated immediately. ________

2.1.8 If radiation monitor 1RE-2565 turns magenta color on the Coms Console, it may be inoperable. Reference Precaution 2.1.9.

IF CVI operability is required when this occurs and 1RE-002 AND 1RE-003 are not BOTH operable, immediately terminate any release or vent in progress, verify Containment Dampers closed and notify chemistry. ________

INITIALS

4.2 RELEASE

> **CAUTION**
>
> No part of the release should be performed until the approved Gaseous Effluent Permit is received in the Control Room.

4.2.1 **Verify** that NO additions have been made to the tank since the sample collection Date/Time listed on the release permit by checking the pressure of the GDT to be released, recorded on Data Sheet 1. ________

4.2.2 **Verify** Chemistry has **performed** a source AND channel check of Waste Gas Processing System Effluent Monitor A-RE-0014. IF A-RE-0014 is operable, **perform** a pulse check A-RE-0014 as follows: ________

a. **Notify** the Control Room to expect an alarm from A-RE-0014 on the Digital Radiation Monitor System. ________

> NOTE
>
> A-HS-0014 (trip indication) white light will remain illuminated until full open.

b. **Open** WASTE GAS DISCHARGE VALVE A-RV-0014 by **performing** the following:

(1) **Set** A-HIC-0014 to 0% DEMAND. ________

(2) **Place** A-HS-0014 in OPEN. ________

(3) **Set** A-HIC-0014 to 100%. ________

(4) **Verify** A-RV-0014 OPENS. ________

c. **Request** Chemistry to **activate** the pulse test on channel A-RE-0014. ________

INITIALS

d. **Verify** the following:

- WASTE GAS DISCHARGE VALVE A-RV-0014 CLOSES. ________

AND

- Hi Radiation alarm in Control Room annunciates. ________

e. **Place** A-HS-0014 in the OPEN position and **check** that A-RV-0014 remains in the closed position. ________

NOTE

After performing a pulse check on A-RV-0014 it is normal that A-RV-0014 will not reopen until Chemistry restores channel A-RV-0014, this action proves the operability of A-RE-0014.

f. **Notify** Chemistry that A-RV-0014 has closed and will NOT reopen. ________

g. **Request** Chemistry to **restore** channel A-RE-0014 to normal. ________

4.2.3 IF operable, **verify** channel check of A-FT-0014 has been **performed** by Chemistry. ________

INITIALS

4.2.4 **Place** WASTE GAS DISCHARGE CONTROL VALVE A-HS-0014 in CLOSE (1-PGPP) (RD-56). ________

4.2.5 **Set** WASTE GAS DISCHARGE CONTROL A-HIC-0014 to 0% demand (1-PGPP) (RD-56). ________

NOTE

IF the Radioactive Gaseous Monitoring Instrument ARE-0014 OR Flow Transmitter AFT-0014 is determined to be inoperable PRIOR to start of the release, the release may be performed IF the requirements of ODCM Section 3.1.1 are met.

Critical

4.2.6 **Perform** the appropriate Checklist to align the Gaseous Waste Processing System for the tank to be released: (CV REQUIRED)

Gas Decay Tank #1-Checklist #1 ________

Gas Decay Tank #2-Checklist #2 ________

Gas Decay Tank #3-Checklist #3 ________

Gas Decay Tank #4-Checklist #4 ________

Gas Decay Tank #5-Checklist #5 ________

Gas Decay Tank #6-Checklist #6 ________

Gas Decay Tank #7-Checklist #7 ________

Shutdown Gas Decay Tank #9-Checklist #8 ________

Shutdown Gas Decay Tank #10-Checklist #9 ________

INITIALS

Critical
4.2.7 **Perform** the appropriate Checklist to align the Gaseous Release Header for the tank to be released:
(CV REQUIRED)

Gas Decay Tank #1-Checklist #1 ________

Gas Decay Tank #2-Checklist #2 ________

Gas Decay Tank #3-Checklist #3 ________

Gas Decay Tank #4-Checklist #4 ________

Gas Decay Tank #5-Checklist #5 ________

Gas Decay Tank #6-Checklist #6 ________

Gas Decay Tank #7-Checklist #7 ________

Shutdown Gas Decay Tank #9-Checklist #8 ________

Shutdown Gas Decay Tank #10-Checklist #9 ________

4.2.8 **Verify** all conditions of the Gaseous Effluent Permit that MUST be satisfied PRIOR to the release are met. ________

Critical
4.2.9 **Verify** the tank aligned for release is the same tank for which the Gaseous Effluent Permit was issued. ________

____
CV

4.2.10 **Note** the maximum allowable release flow rate and A-RE-0014 setpoint given on the Gaseous Effluent Permit. ________

4.2.11 **Notify** the Unit 1 Control Room that the release is starting so they can **monitor** the flow and radiation data. (IPC points F6416 and R6253) ________

4.2.12 **Place** A-HS-0014 (1-PGPP) (RD-56) in OPEN. ________

INITIALS

CAUTION

All Unit 1 and Unit 2 GDTs and SDTs should be monitored to verify that only the GDT or SDT being released is decreasing in pressure.

4.2.13 IF a GDT or SDT NOT being released decreases in pressure, immediately **stop** the release and **notify** the SS. _________

4.2.14 Continuously **monitor** all Gas Decay Tanks and Shutdown Decay Tanks pressures during the first hour of the release, AND THEN **check** all pressures hourly until the release is complete. **Document** on Data Sheet 1. _________

CAUTION

Do not exceed the maximum allowable release rate or A-RE-0014 setpoint stated on the release permit. If at any time during the release, the allowable release rate or A-RE-0014 setpoint is exceeded, the release should be stopped and the SS notified.

4.2.15 IF AFT-0014 is OPERABLE, **monitor** A-RI-0014 and release flow rate while **adjusting** A-HIC-0014 to **obtain** the required release rate. _________

INITIALS

4.2.16 IF AFT-0014 is INOPERABLE, the following steps must be **performed** to **comply** with the action step of the ODCM. **Verify** Flowrate by **performing** the following:

> NOTE
>
> A-HS-0014 (trip indication) white light will remain illuminated until full open.

> **CAUTION**
>
> The Release Permit flow rate should be immediately verified, IF the (trip indication) white light does not remain illuminated until HIC-0014 has fully opened.

a. **Adjust** A-HIC-0014 to 20% OPEN. ________

b. **Log** the start time and initial pressure of the GDT being released in the ABO logbook. ________

c. WHEN the initial pressure has decreased by 2 psig, **perform** the following to **verify** the initial flowrate is within the limits of the release permit:

(1) **Subtract** the present pressure reading from the initial pressure reading then **divide** the result by 14.7. ________

(2) **Multiply** the result of 4.2.16c(1) by 600. ________

(3) **Divide** the result of 4.2.16c(2) by the number of minutes the release has been occurring. The resultant number is the flowrate in standard cubic feet per minute. ________

d. **Adjust** A-HIC-0014 as needed to comply with the permit release rate. ________

e. Every 4 hours, **verify** the release rate by **performing** steps 4.2.16.c(1) thru 4.2.16.c(3). ________

4.2.17 **Record** the start parameters on the Gaseous Effluent Permit. ________

INITIALS

4.2.18 WHEN tank pressure falls below 20 psig, **switch** the associated pressure indicator on 1-PGPP (RD-56) to the LOW RANGE position. ________

4.2.19 WHEN tank pressure falls to 10 psig, or the required amount is released, **terminate** the release as follows:

a. **Place** A-HS-0014 (1-PGPP)(RD-56)in CLOSE. ________

________ IV

b. **Set** A-HIC-0014 (1-PGPP)(RD-56) to 0%. ________

________ IV

c. **Record** the stop time, date, AND final tank pressure on the Gaseous Effluent Permit. ________

WINDOW A08

ORIGIN	**SETPOINT**	
A-RE-0014	Variable	PLANT VENT MONITOR HI RAD.

1.0 **PROBABLE CAUSE**

1. High radiation level in gas being released.
2. Radiation from safety valve release.

2.0 **AUTOMATIC ACTIONS**

Plant Vent Radwaste Gas, A-RV-0014, closes.

Critical
3.0 **INITIAL OPERATOR ACTIONS**

Verify A-RV-0014 is closed

WINDOW A08
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Determine** the gas discharge line radiation using A-RI-0014 on Panel PGPP.

2. **Verify** A-1902-U4-004 closed. ________

   ____
   IV

3. **Notify** the Control Room of the alarm and system status prior to performing the remaining actions.

4. IF planned release was in progress:

   a. **Request** Health Physics to sample waste gas decay shutdown tank being released.

   b. **Check** release conditions and rate.

5. IF release is not planned:

   a. **Check** waste gas decay shutdown tank pressure using A-PIS-1056 and A-PIS-1057 on Panel PGPP.

   b. **Check** status of A-PSV-7884A and A-PSV-7884B.

   c. IF relief valves have lifted at 100 psig, **isolate** the source.

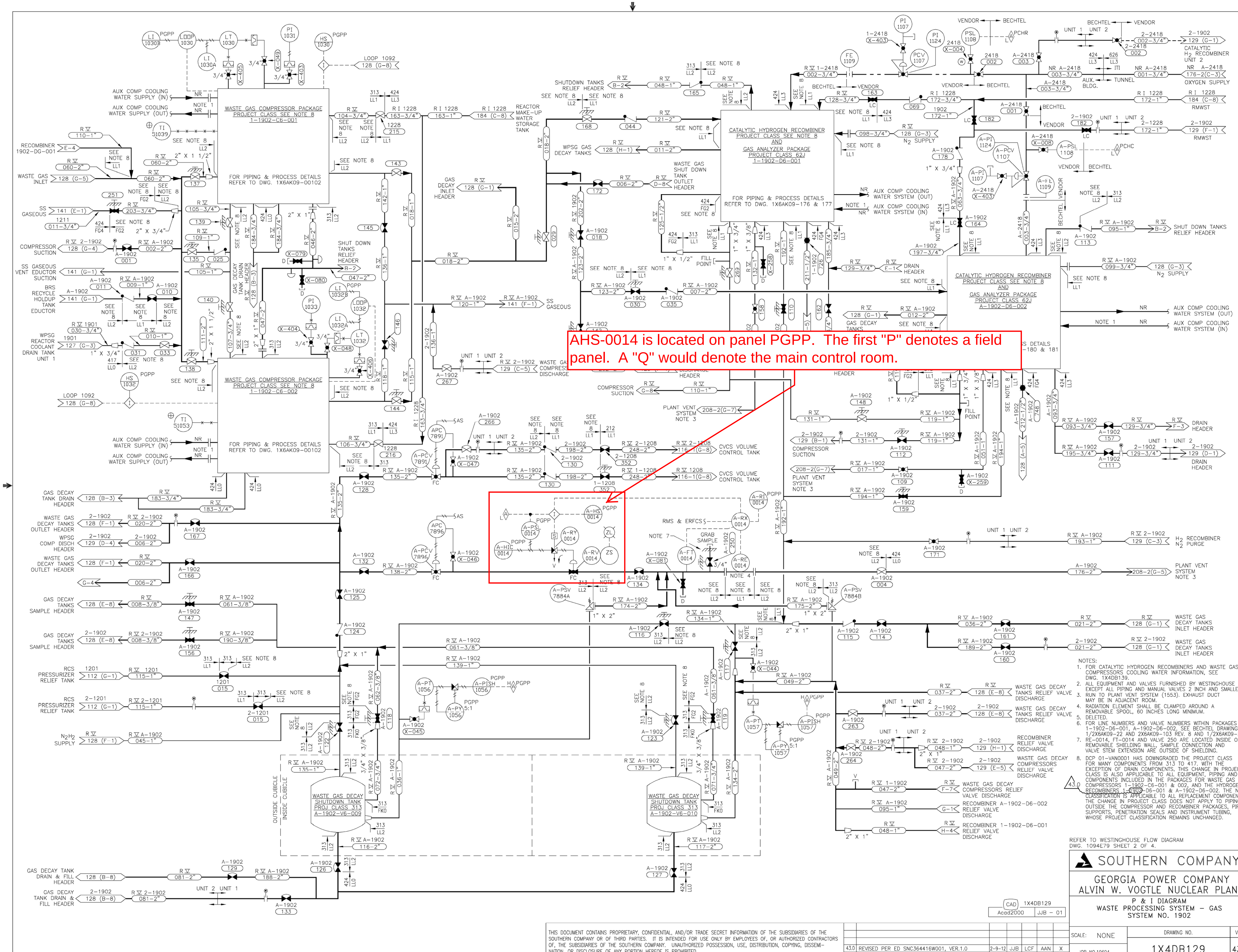
6. **Refer** to the requirements of the ODCM Manual.

7. IF equipment failure is indicated, **initiate** maintenance as required.

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE


AHS-0014 is located on panel PGPP. The first "P" denotes a field panel. A "Q" would denote the main control room.
WASTE GAS COMPRESSOR PACKAGE
PROJECT CLASS SEE NOTE 8
1-1902-C6-001
FOR PIPING & PROCESS DETAILS REFER TO DWG. 1X6AK09-00102
CATALYTIC HYDROGEN RECOMBINER PACKAGE
PROJECT CLASS SEE NOTE 8
AND
GAS ANALYZER PACKAGE
PROJECT CLASS 62J
1-1902-D6-001
FOR PIPING & PROCESS DETAILS REFER TO DWG. 1X6AK09-176 & 177
CATALYTIC HYDROGEN RECOMBINER PACKAGE
PROJECT CLASS SEE NOTE 8
AND
GAS ANALYZER PACKAGE
PROJECT CLASS 62J
A-1902-D6-002
WASTE GAS DECAY SHUTDOWN TANK
PROJ CLASS 313
A-1902-V6-009
WASTE GAS DECAY SHUTDOWN TANK
PROJ CLASS 313
A-1902-V6-010
OUTSIDE CUBICLE
INSIDE CUBICLE
NOTES:
1. FOR CATALYTIC HYDROGEN RECOMBINERS AND WASTE GAS COMPRESSORS COOLING WATER INFORMATION, SEE DWG. 1X4DB139.
2. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND MANUAL VALVES 2 INCH AND SMALLER.
3. RUN TO PLANT VENT SYSTEM (1553). EXHAUST DUCT MAY BE IN ADJACENT ROOM.
4. RADIATION ELEMENT SHALL BE CLAMPED AROUND A REMOVABLE SPOOL, 60 INCHES LONG MINIMUM.
5. DELETED.
6. FOR LINE NUMBERS AND VALVE NUMBERS WITHIN PACKAGES 1-1902-D6-001, A-1902-D6-002, SEE BECHTEL DRAWING 1/2X6AK09-22 AND 2X6AK09-103 REV. 8 AND 1/2X6AK09-21.
7. RE-0014, FT-0014 AND VALVE 250 ARE LOCATED INSIDE OF REMOVABLE SHIELDING WALL, SAMPLE CONNECTION AND VALVE STEM EXTENSION ARE OUTSIDE OF SHIELDING.
8. DCP 01-VAN0001 HAS DOWNGRADED THE PROJECT CLASS FOR MANY COMPONENTS FROM 313 TO 417. WITH THE EXCEPTION OF DRAIN COMPONENTS, THIS CHANGE IN PROJECT CLASS IS ALSO APPLICABLE TO ALL EQUIPMENT, PIPING AND COMPONENTS INCLUDED IN THE PACKAGES FOR WASTE GAS COMPRESSORS 1-1902-C6-001 & 002, AND THE HYDROGEN RECOMBINERS 1-1902-D6-001 & A-1902-D6-002. THE NEW CLASSIFICATION IS APPLICABLE TO ALL REPLACEMENT COMPONENTS. THE CHANGE IN PROJECT CLASS DOES NOT APPLY TO PIPING OUTSIDE THE COMPRESSOR AND RECOMBINER PACKAGES, PIPE SUPPORTS, PENETRATION SEALS AND INSTRUMENT TUBING, WHOSE PROJECT CLASSIFICATION REMAINS UNCHANGED.
43.0
REFER TO WESTINGHOUSE FLOW DIAGRAM DWG. 1094E79 SHEET 2 OF 4.
Q
SOUTHERN COMPANY
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT
P & I DIAGRAM
WASTE PROCESSING SYSTEM - GAS
SYSTEM NO. 1902
CAD 1X4DB129
Acad2000
JJB - 01
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43.0 REVISED PER ED SNC364416W001, VER.1.0 2-9-12 JJB LCF AAN X
NO. VERSIONS DATE DR CHK APPV DTL
SCALE: NONE
JOB NO.10604
DRAWING NO. 1X4DB129
VER. 43.0
SIZE E 34X44
DRAWING CATEGORY: CRITICAL

1. 072K3.02 001/LOIT/RO/M/F 3.1/3.5/072K3.02/LO-TA-32007///072K1.03

Initial conditions:

- Fuel movement is in progress in Unit 1 spent fuel pool.
- ARE-2533A/B, Fuel Handling Building Effluent Radiogas Monitor, is OOS.
- ARE-2532A/B, Fuel Handling Building Effluent Radiogas Monitor, is in service.

Current conditions:

- 1RE-008, Fuel Handling Building Area Monitor, detector fails high.
- Chemistry has removed 1RE-008 from service.

Which one of the following completes the following statement?

The personnel in the SFP area were alerted to the failed detector by a __(1)__,

and

for movement of irradiated fuel in the spent fuel pool, __(2)__ is required to be in service.

	__(1)__	__(2)__
A.	flashing light ONLY	ARE-2532A/B
B.	flashing light ONLY	1RE-008
C.✓	flashing light and audible alarm	ARE-2532A/B
D.	flashing light and audible alarm	1RE-008

K/A

072 Area Radiation Monitoring

K3.02 Knowledge of the effect that a loss or malfunction of the ARM system will have on the following:

- Fuel handling operations.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of how the personnel in the fuel handling building would be alerted to local area monitor RE-008 alarm and the impacts to fuel handling operations.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17100-1, if 1RE-0008 is in high alarm, an audible horn alarm will sound and a

strobe light will illuminate and the Fuel Handling Buidling is required to be evacuated. (Note: Per ARP 17102-1 page 5, not all rad monitors have both a horn and a strobe light.)

Per TRM 13.3.6 Table 13.3.6-1, only 1 Channel of FHB Rad monitor actuation logic is required to be FUNCTIONAL. Since ARE-2532A/B remains in service, the LCO is met. Personnel will be required to leave the FHB when the alarm comes in. Once the issue is addressed and 1RE-0008 removed from service and the alarm silenced, fuel movement may be resumed. There are no requirements to have ARM 1RE-0008 in service during fuel movement. 1RE-0008 is not listed in Tech Specs, TRM, or ODCM.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per ARP 17100-1, if 1RE-0008 is in high alarm, an audible horn alarm will sound and a strobe light will illuminate. However, the fuel handling building is at times a high noise area when the area circulator fans are in service, and other devices such as the CAS crane and fuel handling machine have audible alarms tha could be confused with an alarm horn. It is reasonable for a candidate with insufficient knowledge of 1RE-0008 to conclude that only a strobe light would be suitable for this area.

The second part is correct. Since ARE-2532A/B remains in service, LCOs 13.3.6 AND 13.9.5 are met. There are no requirements in Tech Specs, TRM, or ODCM for 1RE-008 to be functional and therefore it does not impact fuel handing operations in the fuel handling building.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Since ARE-2532A/B remains in service, LCOs 13.3.6 AND 13.9.5 are met. There are no requirements in Tech Specs, TRM, or ODCM for 1RE-008 to be functional and therefore it does not impact fuel handing operations in the fuel handling building. However, TRM 13.3.6 does direct actions to be taken for TRM 13.9.5 if the required channel of actuation logic is nonfunctional. These actions would include either placing a FHB filter in operation or suspending fuel movement. Therefore, it the candidate believes 1RE-008 is part of TRM 13.3.6, it is reasonable for them to believe that fuel movement could not be resumed with it OOS.

C.Correct. The first part is correct. Per ARP 17100-1, if 1RE-0008 is in high alarm, an audible horn alarm will sound and a strobe light will illuminate.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G2
K/A#	072K3.02
Importance Rating:	3.1 / 3.5
Technical Reference:	TRM 13.3.6, Rev 31.0, pages 13.3-19 & 20 TRM 13.9.5, Rev 3.0 9/18/03, pages 13.9-6 & 7 ARP 17100-1, Rev 26.2, page 14 ARP 17102-1, Rev 20.3, page 5
References provided:	None
Learning Objective:	LO-LP-39207-03 For any given item in section 13.3 of the Technical Requirements Manual, be able to: a. State the Technical Requirement (TR) for operation. b. State any one hour or less required actions. LO-TA-32007 Verify Proper Automatic actions to high radiation alarms
Question origin:	MODIFIED - HL18 072K1.03
Cognitive Level:	M/F
10 CFR Part 55 Content:	41.7 / 45.6
Comments:	

You have completed the test!

1. 072K1.03 001/2/2/ARM - FHBI/C/A - 3.6/3.7/MOD - HL-15R AUDIT/HL-18 NRC/RO/SRO/KAJ

A dropped spent fuel assembly in the Unit 1 Spent Fuel Pool has resulted in the following radiation monitor alarms:

Orignal Question

- 1-RE-0008, FHB Area Monitor, indicates HIGH.
- A-RE-2532A(B) and A-RE-2533A(B), FHB Effluent Monitors, indicate ALERT.
- The crew is implementing 18006-C, "Fuel Handling Event".

For the given conditions, which ONE of the following completes the following statement?

1-RE-0008 ___(1)___ provide audible and visual indications of the alarm in the Unit 1 SFP area,

and

the FHB Post-Accident Filtration Units ___(2)___ automatically start.

A. (1) will

(2) will

B.✓ (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

13.3 Instrumentation

TR 13.3.6 Fuel Handling Building Post Accident Ventilation Actuation Instrumentation (common system).

TR 13.3.6 The fuel handling building (FHB) post accident ventilation actuation instrumentation identified in Table 13.3.6-1 shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in either storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required FHB ventilation actuation instruments inoperable.	A.1 Apply Required Actions of TR 13.9.5.	In accordance with TR 13.9.5.

TECHNICAL REQUIREMENT SURVEILLANCES

SURVEILLANCE	FREQUENCY
TRS 13.3.6.1 Perform CHANNEL CHECK	12 hours
TRS 13.3.6.2 Perform COT	18 months
TRS 13.3.6.3 Perform ACTUATION LOGIC TEST	31 days on a STAGGERED TEST BASIS
TRS 13.3.6.4 Perform CHANNEL CALIBRATION	18 months
TRS 13.3.6.5 Perform TADOT	18 months

Table 13.3.6-1

FHB Post Accident Ventilation
Actuation Instrumentation

Instruments	Required Channels	Surveillance Requirements	Trip Setpoint
1. Manual Initiation	1	TRS 13.3.6.5	NA
2. FHB Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	1	TRS 13.3.6.1 TRS 13.3.6.2 TRS 13.3.6.4	(a)
3. Automatic Actuation Logic and Actuation Relays	1	TRS 13.3.6.3	NA

(a) Setpoints will not exceed the limits of TS 5.5.4.g.

13.9 Ref ueling Operations

TR 13.9.5 Fuel Handling Building Post Accident Ventilation System (common system)

TR 13.9.5 Two independent Fuel Handling Building Post Accident Ventilation Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in either storage pool.

--NOTE--
TR 13.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Fuel Handling Building Post Accident Ventilation System inoperable.	A.1 Place the remaining Fuel Handling Building Post Accident Ventilation System in operation discharging through at least one train of HEPA filters and charcoal adsorbers. <u>OR</u>	7 days

(continued)

ACTIONS (continued)

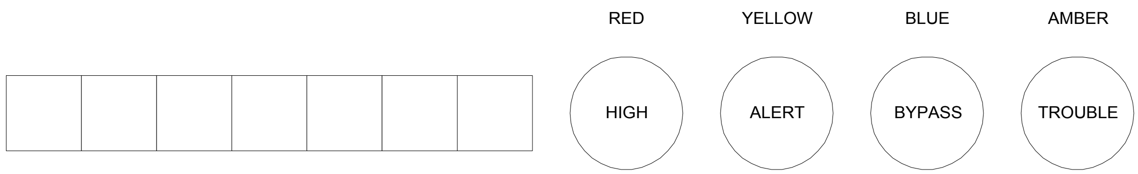
CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Suspend all operations involving movement of irradiated fuel in the fuel handling building, movement of new fuel over irradiated fuel in the fuel handling building, or crane operation with loads over irradiated fuel in the fuel handling building until Required Action A.1 above is met.	7 days
B. Two Fuel Handling Building Post Accident Ventilation Systems inoperable.	B.1	Suspend all operations involving movement of irradiated fuel in the fuel handling building, movement of new fuel over irradiated fuel in the fuel handling building, or crane operation with loads over irradiated fuel in the fuel handling building until Required Action A.1 above is met.	Immediately

TECHNICAL REQUIREMENT SURVEILLANCES

SURVEILLANCE		FREQUENCY
TRS 13.9.5.1	Verify system operation by initiating each system from the control room with flow through the HEPA filters and charcoal adsorbers and operating each system for ≥ 10 continuous hours with the heater circuit energized.	15 days on a STAGGERED TEST BASIS

(continued)

Each channel on the SRDC has a separate display. Normally each display reads the radiation activity level being monitored in 3 digits and an exponent. Units vary from channel to channel. Each channel has an Alert, High and Equipment Trouble alarm display and an indicator that the SRDC is bypassed:



On detecting a high radiation level, the audible alarm on ALB 05 sounds and the red HIGH indicator lamp on the SRDC channel lights. The alarm is also indicated in the TSC and the Health Physics and Chemistry Labs, by displaying the channel identification number on their CRT in red. The alarm is also displayed at the Communications Console (QRM1). A loud horn and a strobe light may announce the high alarm close to the detector.

Alert, Bypass and Equipment Trouble indications do not sound audible alarms.

For very high radiation levels, the TOP OF SCALE, the EQUIPMENT TROUBLE and the HIGH alarms will all light and the sections of the digital display go to "9999999". This causes the alarm to latch, so it will not automatically clear when the radiation level drops. The TOP OF SCALE must be manually reset at the Channel Display and Control Area (CDCA) on the SRDC.

A high alarm will also latch and require a manual resetting.

The Bypass indicates that the channel has been put in the local control mode at the Data Processing Module (DPM) and for 1-RE-0002 or 1-RE-0003 the Containment Ventilation Isolation Block Switches have been placed in the BLOCK position.

Not all rad monitors have audiable horn and strobe light

ORIGIN	SETPOINT	
Area Monitor	As determined by Chemistry Department	1-RE-0008 (High)

> NOTE
>
> For other than HIGH conditions see Pages 4 and 5. ☐

1.0 **PROBABLE CAUSE**

Increase in radiation level near Unit 1 Spent Fuel Pool in the Fuel Handling Building.

2.0 **AUTOMATIC ACTIONS**

On the south wall of the Fuel Handling Building Spent Fuel Pool Room near the door:

a. Alarm horn on 1-RA-0008 sounds.

b. Strobe light on 1-RA-0008 blinks.

3.0 **INITIAL OPERATOR ACTIONS**

Evacuate the Fuel Handling Building.

1. 073K1.01 001/LOIT AND LOCT/RO/C/A 3.6/3.9/073K1.01/LO-PP-04101-03///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- 1RE-1950, ACCW - Liquid, radiation level trends up and then stabilizes.
- ACCW surge tank level increases from 50% to 55% and then stabilizes.

Which one of the following completes the following statement?

A __(1)__ heat exchanger tube leak has occurred,

and

the ACCW surge tank level stabilized due to an automatic isolation of the flow path due to high __(2)__.

	__(1)__	__(2)__
A.	letdown	flow
B.	letdown	temperature
C.✓	thermal barrier	flow
D.	thermal barrier	temperature

K/A

073 Process Radiation Monitoring

K1.01 Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems:

- Those systems served by PRMs

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the physical connection between ACCW and systems that interface with the RCS (specifically the letdown heat exchange and RCP thermal barrier) and the associated responses of the process rad monitors and system response.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17100-1, the source of leakage causing 1RE-1950 to alarm could come from either a letdown heat exchanger or RCP thermal barrier leak. Both RE-1950 and surge

tank level trending up and then stabilizing are symptoms of the leak being isolated. Letdown does not have automatic isolation signals from either RE-1950 or the ACCW system. RCP thermal barriers have both individual outlet auto isolations as well as a common header outlet isolation valves. Per 17004-1 ALB A05, B05, C05, and D05, the individual RCP thermal barrier outlet isolation valves will close at a flow rate of 65 gpm. The ARP directs trending RE-1950 and ACCW surge tank level to validate the alarm, and verify HV-19053 is CLOSED. Per ARP17004-1 B06, the common thermal barrier outlet isolation valve closes on a pressure of 155 psig in the header. (Reference P&ID 1X4DB138-2 for thermal barrier isolation valve and flow transmitter.)

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. As described above in the Explanation of Required Knowledge, the stem gives symptoms of a thermal barrier tube leak and subsequent automatic isolation. However, a letdown heat exchanger leak is also a possible source of leakage per ARP 17100-1 and would exhibit some of the same symptoms (ie. a letdown heat exchanger leak would result in the indication provided in the stem, except indications would contine to rise rather than stablize).

The second part is correct. Per ARP 17004-1, the individual thermal barrier return lines isolate automatically on high flow. The common isolation will isolate on high flow or pressure. This distractor is plausible since the letdown flow path has automatic isolations based on temperature and has several alarms associated with flow and temperature.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per ARP 17004-1, the individual thermal barrier return lines isolate automatically on high flow. However, letdown does have an isolation on high temperature. (Reference P&ID 1X4DB114 for valves HV15214 and HV8160.)

C. Correct. The first part is correct. As described above in the Explanation of Required Knowledge, the stem gives symptoms of a thermal barrier tube leak and subsequent automatic isolation.

The second part is correct. See the first part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the first part of choice B above.

Level:	RO
Tier # / Group #	T / G
K/A#	073K1.01
Importance Rating:	3.6 / 3.9

Technical Reference: ARP 17004-1, Rev 24.0, pages 3, 21 thru 26
ARP 17100-1, Rev 26.2, pages 50 thru 52
ELEMENTARY 1X3D-BD-C01R, Rev 3.0
P&ID 1X4DB114, Rev 41.0
P&ID 1X4DB138-2, Rev 19.0

References provided: None

Learning Objective: LO-PP-04101-04 From memory describe the expected system response and operator corrective actions for each of the following:

g. Thermal barrier heat exchanger leak
i. CVCS letdown heat exchanger leak

LO-PP-04101-03 Describe how ACCW surge tank level and RE-1950 are used to determine source of in-leakage and when the in-leakage is isolated.

Question origin: BANK

Cognitive Level: C/A

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments:

You have completed the test!

ALB-04

	(1)	(2)	(3)	(4)	(5)	(6)
A	ACCW SURGE TK HI/LO LVL	ACCW LO HDR PRESS	ACCW RCP 1 CLR LO FLOW	ACCW RCP 1 CLR OUTLET HI TEMP	ACCW RCP 1 THRM BARRIER HX HI FLOW	
B	BOP PROT GR I BYPASS	ACCW RX COOLANT DRN TK HX LO FLOW	ACCW RCP 2 CLR LO FLOW	ACCW RCP 2 CLR OUTLET HI TEMP	ACCW RCP 2 THRM BARRIER HX HI FLOW	ACCW RCP THRM BARRIER HI PRESS
C	BOP PROT GR II BYPASS	ACCW EXCESS LTDN HX LO FLOW	ACCW RCP 3 CLR LO FLOW	ACCW RCP 3 CLR OUTLET HI TEMP	ACCW RCP 3 THRM BARRIER HX HI FLOW	
D	BOP PROT GR III BYPASS	ACCW RTN HDR FROM RCP LO FLOW	ACCW RCP 4 CLR LO FLOW	ACCW RCP 4 CLR OUTLET HI TEMP	ACCW RCP 4 THRM BARRIER HX HI FLOW	
E	TRAIN A SYS STATUS MON PNL ALERT	TRAIN B SYS STATUS MON PNL ALERT		BOP PROCESS PROT CABINET DOORS OPEN	BOP PROCESS PROT DOOR OPEN >1 CABINET	BOP PCS CABS PWR SUPPLY FAILURE
F	TRAIN C SYS STATUS MON PNL ALERT		TRAIN A SHUTDOWN PNL ON LOCAL CNTL	TRAIN B SHUTDOWN PNL ON LOCAL CNTL	TRAIN C SHUTDOWN PNL ON LOCAL CNTL	“ASIS” TROUBLE

WINDOW B05

ORIGIN	SETPOINT	ACCW RCP 2 THRM BARRIER HX HI FLOW
1-FSH-19054B	65 gpm	

1.0 **PROBABLE CAUSE**

1. Leak into Auxiliary Component Cooling Water (ACCW) from Reactor Coolant Pump (RCP) 2 Thermal Barrier Heat Exchanger (HX).

2. Leak downstream of 1-FE-19054.

2.0 **AUTOMATIC ACTIONS**

RCP 2 Thermal Barrier HX Outlet Isolation Valve 1-HV-19053 closes on high flow of 69 gpm.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Verify** RCP 2 seal injection and return flows are normal.

2. **Check** Radiation Monitor 1-RE-1950 for signs of radiation leakage into ACCW.

3. **Check** computer point L2700 for increasing Surge Tank level.

   a. IF ACCW Surge tank level was NOT rising,

      (1) IF closure was spurious (for example closed during ACCW Pump swap, power supply fluctuation or maintenance activities) AND leakage is NOT suspected) m**onitor** Surge tank level AND attempt to **open** 1-HV-19053.

      (2) IF closure reason is unknown **OR** leakage is suspected, **maintain** 1-HV-19053 closed, continue with this procedure AND **initiate** maintenance.

   b. IF ACCW Surge tank level was **OR** is rising, **verify** 1-HV-19053 CLOSED.

WINDOW B05
(Continued)

4. IF RCP 2 temperatures and seal flows are normal and there is no indication of radiation in the ACCW System, **initiate** maintenance as required to correct cause of the alarm.

5. IF 1-HV-19053 closes, **verify** the RCP is operated within the limits established in 13003-1, "Reactor Coolant Pump Operation" and **ensure** actions of Technical Requirement (TR) 13.7.4 are met.

6. IF 1-HV-2041 closed, **restore** flow to the intact thermal barriers as follows:

   a. **Check** computer trend (IPC Point L2700) to determine if ACCW Surge tank level was rising prior to isolation.

   b. **Check** ACCW Surge tank level is now stable (leak is isolated).

   c. **Check** leaking thermal barrier isolated (1-HV-19053 closed).

   d. **Check** 1-HV-2041 has not been closed for greater than 30 minutes to preclude the possible formation of steam in the thermal barrier.

      (1) IF 1-HV-2041 has been closed greater than 30 minutes, **contact** engineering to evaluate opening 1-HV-2041 PRIOR to proceeding with this step.

   e. **Monitor** ACCW Surge tank for level rise in the following step:

      (1) **Open** 1-HV-2041.

      (2) IF a level rise is observed, **close** 1-HV-2041.

      (3) **Check** Surge Tank level stable.

7. IF it becomes necessary to stop the pump and:

   a. Reactor power is above 15%:

      (1) **Trip** the reactor and initiate 19000-C, "E-0 Reactor Trip Or Safety Injection,"

      (2) **Stop** the pump.

   b. Reactor power is 15% or less, **stop** the pump and initiate 18005-C, "Partial Loss Of Flow."

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB138-2, 1X3D-BD-L03L, CX5DT101-130

WINDOW B06

ORIGIN	SETPOINT	
1-PSH-2041B	155 psig	ACCW RCP THRM BARRIER HI PRESS

1.0 **PROBABLE CAUSE**

Leak into Auxiliary Component Cooling Water (ACCW) from a Reactor Coolant Pump (RCP) Thermal Barrier Heat Exchanger (HX).

2.0 **AUTOMATIC ACTIONS**

ACCW RCP Thermal Barrier Outlet Header Isolation Valve 1-HV-2041 closes on high header pressure and/or flow.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Verify** valve 1-HV-2041 is closed.
2. **Verify** all RCP seal injection and return flows are normal.
3. **Check** Radiation Monitor 1-RE-1950 for signs of radiation leakage into ACCW.
4. **Verify** the RCPs are operated within the limits established in 13003-1, "Reactor Coolant Pump Operation."

WINDOW B06
(Continued)

5. IF 1-HV-2041 closed, **restore** flow to the intact thermal barriers as follows:

a. **Check** computer trend (IPC Point L2700) to determine if ACCW Surge tank level was rising prior to isolation.

(1) IF ACCW Surge tank level was not rising, THEN perform (a) **OR** (b):

(a) IF closure was spurious (for example closed during ACCW Pump swap, power supply fluctuation or maintenance activities) AND leakage is NOT suspected) **monitor** Surge tank level AND attempt to **open** 1-HV-2041.

(b) IF closure reason is unknown, **maintain** 1-HV-2041 closed, **initiate** maintenance AND go to step 5.b.

(2) IF ACCW Surge tank level was rising, **continue** with next step.

b. **Check** ACCW Surge tank level is now stable.

c. **Check** 1-HV-2041 has not been closed for greater than 30 minutes to preclude the possible formation of steam in the thermal barrier.

(1) IF 1-HV-2041 has been closed greater than 30 minutes, **contact** engineering to evaluate opening 1-HV-2041 PRIOR to proceeding with this step.

d. **Check** ANY individual RCP Isolation valve CLOSED to isolate thermal barrier leaking.

- 1-HV-19051 for RCP-1
- 1-HV-19053 for RCP-2
- 1-HV-19055 for RCP-3
- 1-HV-19057 for RCP-4

e. **Monitor** ACCW Surge tank for level rise in the following step:

(1) **Open** 1-HV-2041.

(2) IF a level rise is observed, **close** 1-HV-2041.

(3) **Check** Surge Tank level stable.

WINDOW B06
(Continued)

6. IF it becomes necessary to stop the pump and:

   a. Reactor power is above 15%:

      (1) **Trip** the reactor,

      (2) **Initiate** 19000-C, "E-0 Reactor Trip Or Safety Injection,"

      (3) **Stop** affected RCP, and

      (4) **Verify** Thermal Barrier Outlet Isolation Valve of affected RCP is closed.

         - 1-HV-19051 for RCP-1
         - 1-HV-19053 for RCP-2
         - 1-HV-19055 for RCP-3
         - 1-HV-19057 for RCP-4

   b. Reactor power is 15% or less:

      (1) **Stop** affected RCP,

      (2) **Verify** Thermal Barrier Outlet Isolation Valve of affected RCP is closed.

      (3) **Initiate** 18005-C, "Partial Loss Of Flow."

7. IF equipment failure is indicated, **initiate** maintenance as required and **ensure** actions of Technical Requirement (TR) 13.7.4 are met.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB138-2, 1X3D-BD-L03P, CX5DT101-129

ORIGIN	SETPOINT	
Skid mounted Liquid Process Monitor	As determined by Chemistry Department	1-RE-1950 (High)

NOTE

For other than HIGH conditions see Pages 4 and 5. ☐

1.0 PROBABLE CAUSE

High radiation level in the ACCW from inleakage of radioactive water.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Request** Chemistry to sample and analyze the ACCW.
2. **Notify** Health Physics of the alarm.
3. **Locate** the source of inleakage.
   a. **Check** IPC points T0145, P0135 and F0134 (IPC Group 21) for changes, in an attempt to determine if a Letdown HX tube leak.
   b. **Check** IPC points T2714, T2716, T2718 and T2720 (ICP Group 242) for changes, in an attempt to determine if leakage is from a RCP thermal barrier.
4. **Isolate** the source if possible.

1-RE-1950
(Continued)

5. IF 1-RE-1950 is reading high due to Letdown Heat Exchanger Tube leakage:

Critical

a. **Place** LETDOWN TO DEMIN/VCT 1-TV-0129 to the VCT position using 1-HS-0129.

_________ Initial

_________ CV Initial

Critical

(1) **Verify** 1-TV-0129 aligns to the VCT.

_________ Initial

_________ CV Initial

Critical

b. **Place** VCT HUT LETDOWN DIVERT 1-LV-0112A to the HUT position using 1-HS-0112A.

_________ Initial

_________ CV Initial

Critical

(1) **Verify** 1-LV-0112A aligns to the RHUT.

_________ Initial

_________ CV Initial

1-RE-1950
(Continued)

Critical

c. **Isolate** letdown. **Verify** closed:

_________ Initial

_________ CV Initial

(1) 1-HV-8149A, B, C.

(2) 1-LV-0459.

(3) 1-LV-0460.

(4) 1-HV-8152.

(5) 1-HV-8160.

(6) 1-PV-0131,-set to max pressure.

(7) 1-TV-0130,-set to max temperature.

d. **Shut** Letdown Heat Exchanger manual valves:

(1) (AB-A08) 1-1208-U6-041.

(2) (AB-A17) 1-1217-U4-126.

(3) (AB-108) 1-1217-U4-129.

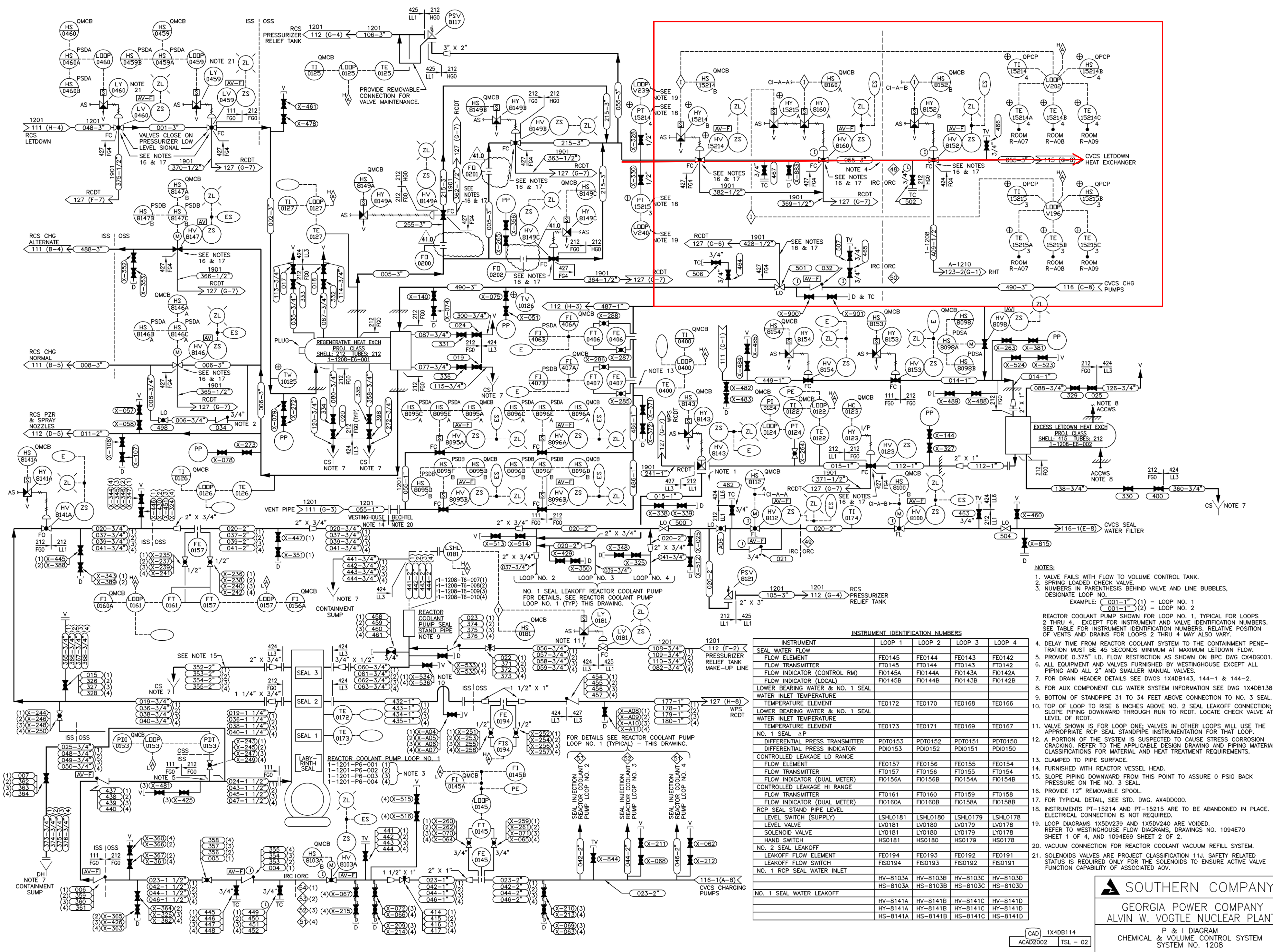
e. **Notify** Chemistry,

f. **Initiate** 18007-C, "CVCS Malfunction" to deal with the loss of letdown.

6. **Obtain** detector trend from the IPC computer.

7. **Monitor** the channel for further changes.

8. IF sampling and analysis determine the channel has malfunctioned, **request** Chemistry to deactivate the channel.



INSTRUMENT IDENTIFICATION NUMBERS

INSTRUMENT	LOOP 1	LOOP 2	LOOP 3	LOOP 4
SEAL WATER FLOW				
FLOW ELEMENT	FE0145	FE0144	FE0143	FE0142
FLOW TRANSMITTER	FT0145	FT0144	FT0143	FT0142
FLOW INDICATOR (CONTROL RM)	FI0145A	FI0144A	FI0143A	FI0142A
FLOW INDICATOR (LOCAL)	FI0145B	FI0144B	FI0143B	FI0142B
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0172	TE0170	TE0168	TE0166
LOWER BEARING WATER & NO. 1 SEAL				
WATER INLET TEMPERATURE				
TEMPERATURE ELEMENT	TE0173	TE0171	TE0169	TE0167
NO. 1 SEAL ΔP				
DIFFERENTIAL PRESS TRANSMITTER	PDT0153	PDT0152	PDT0151	PDT0150
DIFFERENTIAL PRESS INDICATOR	PDI0153	PDI0152	PDI0151	PDI0150
CONTROLLED LEAKAGE LO RANGE				
FLOW ELEMENT	FE0157	FE0156	FE0155	FE0154
FLOW TRANSMITTER	FT0157	FT0156	FT0155	FT0154
FLOW INDICATOR (DUAL METER)	FI0156A	FI0156B	FI0154A	FI0154B
CONTROLLED LEAKAGE HI RANGE				
FLOW TRANSMITTER	FT0161	FT0160	FT0159	FT0158
FLOW INDICATOR (DUAL METER)	FI0160A	FI0160B	FI0158A	FI0158B
RCP SEAL STAND PIPE LEVEL				
LEVEL SWITCH (SUPPLY)	LSHL0181	LSHL0180	LSHL0179	LSHL0178
LEVEL VALVE	LV0181	LV0180	LV0179	LV0178
SOLENOID VALVE	LY0181	LY0180	LY0179	LY0178
HAND SWITCH	HS0181	HS0180	HS0179	HS0178
NO. 2 SEAL LEAKOFF				
LEAKOFF FLOW ELEMENT	FE0194	FE0193	FE0192	FE0191
LEAKOFF FLOW SWITCH	FIS0194	FIS0193	FIS0192	FIS0191
NO. 1 RCP SEAL WATER INLET				
	HV-8103A	HV-8103B	HV-8103C	HV-8103D
	HS-8103A	HS-8103B	HS-8103C	HS-8103D
NO. 1 SEAL WATER LEAKOFF				
	HV-8141A	HV-8141B	HV-8141C	HV-8141D
	HY-8141A	HY-8141B	HY-8141C	HY-8141D
	HS-8141A	HS-8141B	HS-8141C	HS-8141D

NOTES:

1. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK.
2. SPRING LOADED CHECK VALVE.
3. NUMBERS IN PARENTHESIS BEHIND VALVE AND LINE BUBBLES, DESIGNATE LOOP NO.
   EXAMPLE: 001-1" (1) - LOOP NO. 1
   001-1" (2) - LOOP NO. 2
   REACTOR COOLANT PUMP SHOWN FOR LOOP NO. 1, TYPICAL FOR LOOPS 2 THRU 4, EXCEPT FOR INSTRUMENT AND VALVE IDENTIFICATION NUMBERS. SEE TABLE FOR INSTRUMENT IDENTIFICATION NUMBERS. RELATIVE POSITION OF VENTS AND DRAINS FOR LOOPS 2 THRU 4 MAY ALSO VARY.
4. DELAY TIME FROM REACTOR COOLANT SYSTEM TO THE CONTAINMENT PENETRATION MUST BE 45 SECONDS MINIMUM AT MAXIMUM LETDOWN FLOW.
5. PROVIDE 0.375" I.D. FLOW RESTRICTION AS SHOWN ON BPC DWG CX4DG001.
6. ALL EQUIPMENT AND VALVES FURNISHED BY WESTINGHOUSE EXCEPT ALL PIPING AND ALL 2" AND SMALLER MANUAL VALVES.
7. FOR DRAIN HEADER DETAILS SEE DWGS 1X4DB143, 144-1 & 144-2.
8. FOR AUX COMPONENT CLG WATER SYSTEM INFORMATION SEE DWG 1X4DB138-2.
9. BOTTOM OF STANDPIPE 31 TO 34 FEET ABOVE CONNECTION TO NO. 3 SEAL.
10. TOP OF LOOP TO RISE 6 INCHES ABOVE NO. 2 SEAL LEAKOFF CONNECTION; SLOPE PIPING DOWNWARD THROUGH RUN TO RCDT. LOCATE CHECK VALVE AT LEVEL OF RCDT.
11. VALVE SHOWN IS FOR LOOP ONE; VALVES IN OTHER LOOPS WILL USE THE APPROPRIATE RCP SEAL STANDPIPE INSTRUMENTATION FOR THAT LOOP.
12. A PORTION OF THE SYSTEM IS SUSPECTED TO CAUSE STRESS CORROSION CRACKING. REFER TO THE APPLICABLE DESIGN DRAWING AND PIPING MATERIAL CLASSIFICATIONS FOR MATERIAL AND HEAT TREATMENT REQUIREMENTS.
13. CLAMPED TO PIPE SURFACE.
14. FURNISHED WITH REACTOR VESSEL HEAD.
15. SLOPE PIPING DOWNWARD FROM THIS POINT TO ASSURE 0 PSIG BACK PRESSURE ON THE NO. 3 SEAL.
16. PROVIDE 12" REMOVABLE SPOOL.
17. FOR TYPICAL DETAIL, SEE STD. DWG. AX4DD000.
18. INSTRUMENTS PT-15214 AND PT-15215 ARE TO BE ABANDONED IN PLACE. ELECTRICAL CONNECTION IS NOT REQUIRED.
19. LOOP DIAGRAMS 1X5DV239 AND 1X5DV240 ARE VOIDED. REFER TO WESTINGHOUSE FLOW DIAGRAMS, DRAWINGS NO. 1094E70 SHEET 1 OF 4, AND 1094E69 SHEET 2 OF 2.
20. VACUUM CONNECTION FOR REACTOR COOLANT VACUUM REFILL SYSTEM.
21. SOLENOIDS VALVES ARE PROJECT CLASSIFICATION 11J. SAFETY RELATED STATUS IS REQUIRED ONLY FOR THE SOLENOIDS TO ENSURE ACTIVE VALVE FUNCTION CAPABILITY OF ASSOCIATED AOV.

Q

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SYSTEM NO. 1208

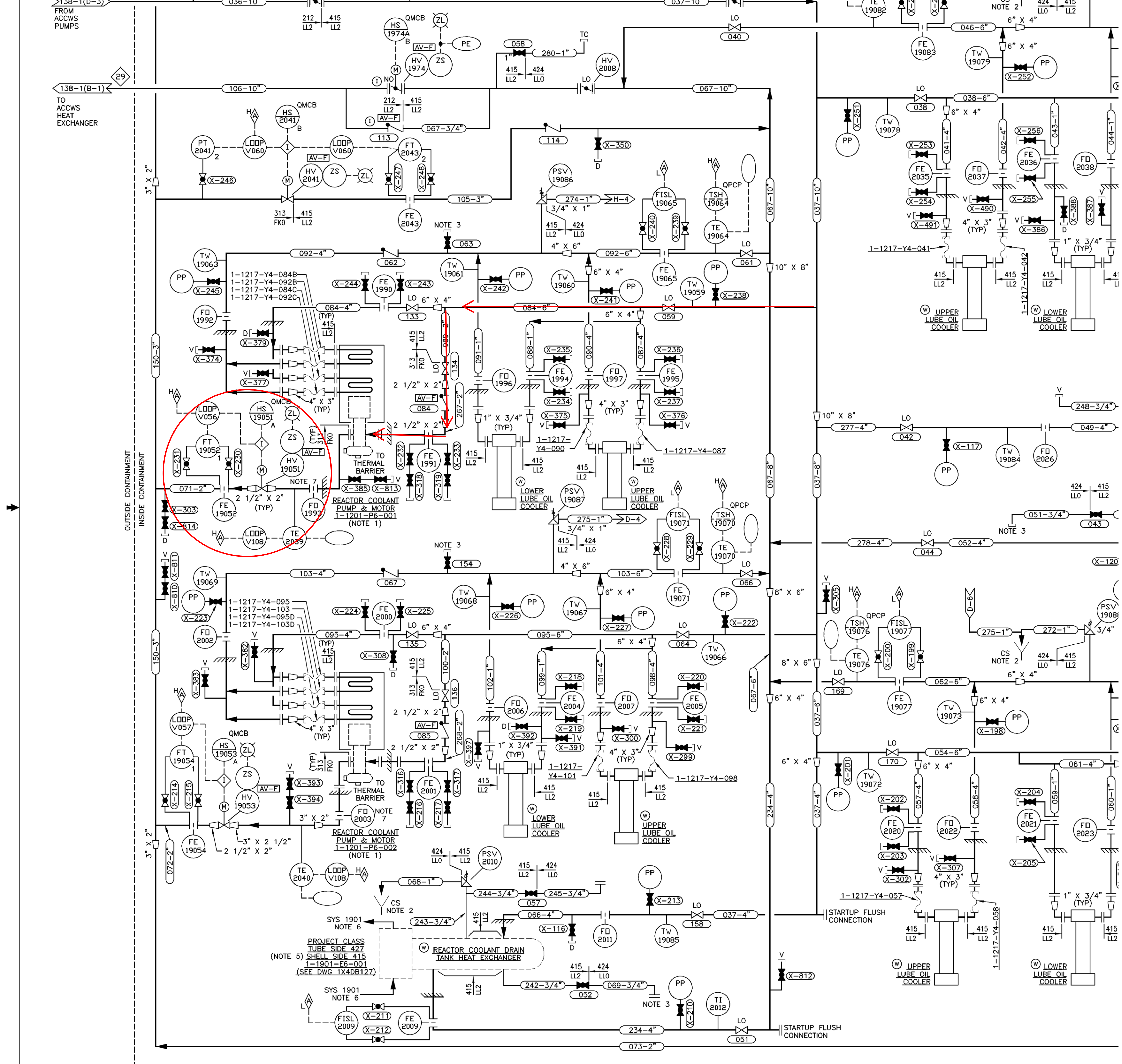
CAD 1X4DB114
ACAD2002 | TSL - 02

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
41.0	REVISED PER ABN 1090006501M002, VER. 1.0	10/07/08	TSL	MWD	JMR	X

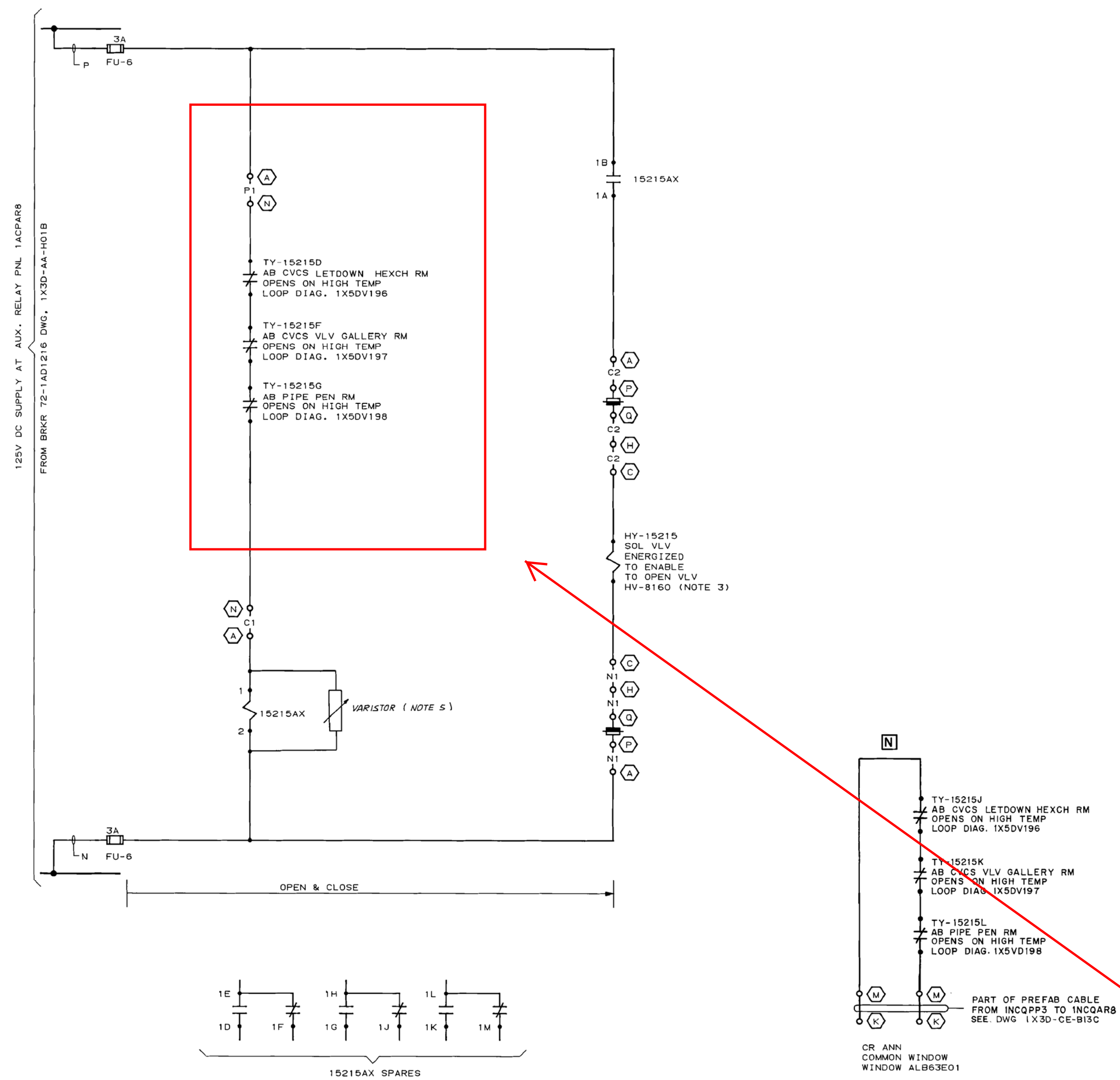
SCALE: NONE | JOB NO.10604 | DRAWING NO. 1X4DB114 | VER. 41.0

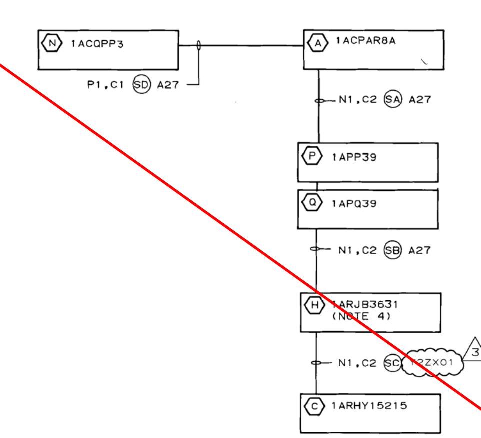
SIZE E 34X44

1 DRAWING CATEGORY: CRITICAL



FROM ACCWS PUMPS
TO ACCWS HEAT EXCHANGER
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
REACTOR COOLANT PUMP & MOTOR 1-1201-P6-001 (NOTE 1)
REACTOR COOLANT PUMP & MOTOR 1-1201-P6-002 (NOTE 1)
TO THERMAL BARRIER
LOWER LUBE OIL COOLER
UPPER LUBE OIL COOLER
REACTOR COOLANT DRAIN TANK HEAT EXCHANGER
PROJECT CLASS TUBE SIDE 427 SHELL SIDE 415 1-1901-E6-001 (SEE DWG 1X4DB127)
(NOTE 5)
SYS 1901 NOTE 6
STARTUP FLUSH CONNECTION
1-1217-Y4-084B
1-1217-Y4-092B
1-1217-Y4-084C
1-1217-Y4-092C
1-1217-Y4-095
1-1217-Y4-103
1-1217-Y4-095D
1-1217-Y4-103D
1-1217-Y4-087
1-1217-Y4-090
1-1217-Y4-098
1-1217-Y4-101
1-1217-Y4-041
1-1217-Y4-042
1-1217-Y4-057
1-1217-Y4-058
138-1(B-1)
138-1(D-5)





NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DRAWINGS LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. ANOTHER SOLENOID VALVE HY-8160 SHALL BE ENERGIZED TO ENABLE TO OPEN VALVE HV-8160 AND DE-ENERGIZED TO CLOSE. SEE 1X3D-BD-C04A.
4. JUNCTION BOX 1ARJB3631 IS ALSO SHOWN ON DWG. 1X3D-BD-C04A.
5. *VARISTOR. GE TYPE Z MODEL V180ZA10 .*

NUCLEAR SAFETY RELATED A

LETDOWN HEXCH INLET PIPE BREAK RM PROTECTION. Q

DEVICE	MFR	TYPE	MODEL NO	REFERENCE DWGS		
				NUMBER	DESCRIPTION	REV
15215AX	P & B	MDR	MDR5061	1X4DB114	P & I DIAGRAM	—
				N/A	ONE LINE DIAGRAM	—
				1X5DN151-2	LOGIC DIAGRAM	—
				N/A	BECHTEL V/P	—

SCHEME 1AR15215
1HY15215
S.U.NO.BG

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
3	INCORP. PER DCP 98-V1N0066	03-25-02	JGM	TSL	MWD	X

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
CHEMICAL & VOLUME CONTROL SYS.
1HY-15215

SCALE: NONE	DRAWING NO.	REV.
JOB NO.10604	1X3D-BD-C01R	3

SIZE E 34X44

1. 076K2.04 001/LOIT/RO/C/A 2.5/2.6/076K2.04///003K2.02/

Initial conditions:

- Unit 1 is at 100% reactor power.
- ACCW pump #1 is running.
- ACCW pump #2 is in standby.

Current conditions:

- An SI occurred and **has** been reset.
- 5 minutes later, RAT '1A' experiences a fault.
- DG1A re-energizes 1AA02 and completes the load sequence.

Which one of the following completes the following statement?

Based on the given sequence of events, ACCW pump #1 __(1)__ be running,

and

ACCW pump #2 __(2)__ be running.

	__(1)__	__(2)__
A.✔	will	will
B.	will	will NOT
C.	will NOT	will
D.	will NOT	will NOT

K/A

076 **Service Water**

K2.04 **Knowledge of bus power supplies to the following:**

- Reactor building closed cooling water.

K/A MATCH ANALYSIS

The question tests the candidate knowledge of which ACCW pump(s) will be in service following various events which change the power status of both 1E 4160V buses.

(Note: Vogtle's bus scheme is extremely simple with just a single 4160V bus powered by a RAT or a DG. Interaction with the Sequencer has been incorporated to increase LOD above a 1.0.)

EXPLANATION OF REQUIRED KNOWLEDGE

Per ELEMENTARY 1X3D-BD-L03B, if an SI actuation occurs, SSPS energizes a slave relay contact in the ACCW pump start circuit that prevents all autostarts. This block will remain until SI has been RESET. The SI actuation does not change the status of either pump beyond inserting this block. Since the SI signal has been reset in this question, the block is no longer inserted and the ACCW pumps will behave normally. When the LOSP occurs on 1AA02, the ACCW header pressure lowers and ACCW Pump #2 will auto start. In parallel with this, the 'A' train sequencer performs a load shed, the already running DG (which started on the SI signal and continues to run until manually shut down) output breaker closes, the bus re-energizes, and the LOSP sequence is run. ACCW Pump #1 is started as part of the LOSP sequence. Therefore, ACCW Pumps #1 and #2 will be running. (Reference ONELINE 1X3D-AA-K02A)

ANSWER / DISTRACTOR ANALYSIS

A. Correct — The first part is correct. ACCW Pump #1 will restart on the LOSP sequence since the SI signal has been reset.

The second part is correct. ACCW Pump #2 will autostart on low header pressure since it is not blocked by the SI signal when ACCW Pump #1 loses power.

B. Incorrect. Plausible. — The first part is correct. See the first part of choice A above.

The second part is incorrect. ACCW Pump #2 will autostart on low header pressure since it is not blocked by the SI signal when ACCW Pump #1 loses power. However, if the candidate does not remember the low header pressure autostart, they will not expect ACCW Pump #2 to be running, but will expect ACCW Pump #1 to be started by the LOSP sequence.

C. Incorrect. Plausible. — The first part is incorrect. ACCW Pump #1 will restart on on the LOSP sequence since the SI signal has been reset. However, if the candidate misses that the SI has been reset, and believes the low pressure autostart is not affected by the SI signal then they will not expect ACCW Pump #1 to start as part of the SI Sequence, but will expect ACCW Pump #2 to have started when ACCW Pump #1 lost power.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. — Both parts are incorrect. ACCW Pump #1 will restart on the LOSP sequence since the SI signal has been reset. ACCW Pump #2 will autostart on low header pressure since it is not blocked by the SI signal when ACCW Pump #1 loses power. However, if the candidate misses the SI reset and believes the SI signal is still present, then neither ACCW pump will be running and this would be the correct answer.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	076K2.04
Importance Rating:	2.5 / 2.6

Technical Reference: ELEMENTARY 1X3D-BD-L03B, Rev 10.0
ONELINE 1X3D-AA-K02A, Rev 15.0

References provided: None

Learning Objective:

LO-LP-60318-01	Describe how the ACCW pumps are affected by a simultaneous loss of offsite power and safety injection.
LO-PP-11101-54	Describe the Operation of the sequencer relative to the EDG.
LO-TA-60005	Respond to a Loss of ACCW per 18022-C
LO-TA-60009A	Respond to a Loss of Class 1E Electrical Systems per 18031-1/2

Question origin: MODIFIED - HL17 NRC # 003K2.02

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7

Comments:

You have completed the test!

1. 003K2.02 001/2/1/RCP-POWER ACCW/H-2.5*/2.6*/NEW/HL17 NRC/RO/SRO/TNT/GCW

Initial conditions:

Original Question

- ACCW pump # 1 is running.
- ACCW pump # 2 is in standby.

Current sequence of events:

- Safety Injection occurs.
- 2 minutes later, RAT "1A" experiences a fault.
- DG1A energizes 1AA02.

- SI has NOT been reset.

Based on the current conditions, which one of the following correctly describes the status of the ACCW pumps after the DG1A load sequence is complete, if no operator actions occur?

	ACCW Pump # 1	ACCW Pump # 2
A.✓	OFF	OFF
B.	RUNNING	OFF
C.	OFF	RUNNING
D.	RUNNING	RUNNING

"M" denotes a manual start.

EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION	PROJECT CLASSIFICATION (ELECTRICAL SYSTEM)	PROCUREMENT SPECIFICATION	BUS NUMBER	BUS VOLTAGE	NAMEPLATE RATING HP (OR KW)	REFERENCE DWG. 1X3D-	LOSS OF COOLANT ACCIDENT SI PREFERRED OFFSITE SOURCE AVAILABLE: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START (NOTE 7)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER & SUBSEQUENT SI: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.b)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.c)	RUNNING TIME AFTER START (NOTE 4)
11202P4001M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04A	S	25.5	C	S	25.5	C	S	25.5	C
11202P4003M01	NUC SERV CLG WTR PP	11E	X4	1AA02	4160V	700	BD-K04C	S	25.5	C	S	25.5	C	S	25.5	C
11202P4005M01	NUC SERV CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	700	BD-K04E	NOTE. 5	30.5	—	NOTE 5	30.5	—	NOTE 5	30.5	—
11203P4001M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01A	S	20.5	C	S	20.5	C	S	20.5	C
11203P4003M01	COMP CLG WTR PP	11E	X4	1AA02	4160V	300	BD-L01C	S	20.5	C	S	20.5	C	S	20.5	C
11203P4005M01	COMP CLG WTR PP (SPARE)	11E	X4	1AA02	4160V	300	BD-L01E	NOTE 5	25.5	—	NOTE 5	25.5	—	NOTE 5	25.5	—
11204P6003M01	SAFETY INJ PP	11E	X6	1AA02	4160V	450	BD-D01C	S	5.5	C	S	5.5	C	M	>30.5	C
11205P6001M01	RESID HT REMVL PP	11E	X6	1AA02	4160V	400	BD-E01A	S	10.5	C	S	10.5	C	M	>30.5	C
11206P6001M01	CONTAINMENT SPRAY PP	11E	X6	1AA02	4160V	400	BD-J01A	NOTE 6	15.5	C	NOTE 6	15.5	C	M	>30.5	C
11208P6002M01	CENTRIFUGAL CHARG PP	11E	X6	1AA02	4160V	600 **	BD-C01A	S	0.5	C	S	0.5	C	S	0.5	C
11217P4001M01	AUX COMP CLG WTR PP	11E	X4	1AA02	4160V	600	BD-L03A	M	>30.5	C	M	>30.5	C	S	15.5	C
11302P4003M01	AUX FDWTR PP-MOTOR DRIVEN	11E	X4	1AA02	4160V	900	BC-F04A	S	20.5	C	S	20.5	C	S	20.5	C
11592C7001M01	ESF CHILLER (COMPRESSOR)	11E	X4	1AA02	4160V	400	BG-G02A	P	>30.5	C	P	>30.5	C	P	>30.5	C
11805S3B04	LOAD CENTER TRANS. 1AB04	11E	X3	1AB04	4160 -480V	1000KVA	BA-D02F	—	—	C	—	0	C	—	0	C
	SECONDARY BREAKER 1AB04	11E	X3	1AB04	480V	—	BA-E02A	—	—	C	S	0.5	C	S	0.5	C
11501A7001M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01A	NOTE 10	NA	NA	NOTE 10	NA	NA	S NOTE 15	50.5	C
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03F	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
11501A7002M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01B	NOTE 10	NA	NA	NOTE 10	NA	NA	S NOTE 15	50.5	C
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03G	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
11501A7005M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01E	NOTE 10	NA	NA	NOTE 10	NA	NA	S	30.5	C
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03K	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
11501A7006M01	CTMT CLG UNIT HIGH SPEED	11E	X4	1AB04	480V	125	BG-B01F	NOTE 10	NA	NA	NOTE 10	NA	NA	S	30.5	C
	CTMT CLG UNIT LOW SPEED	11E	X4	1AB04	480V	62.5	BG-B03L	S	30.5	C	S	30.5	C	NOTE 10	NA	NA
11805S3B05	LOAD CENTER TRANS. 1AB05	11E	X3	1AB05	4.16 -480V	1000KVA	BA-D02L	—	—	C	—	0	C	—	0	C
	SECONDARY BREAKER 1AB05	11E	X3	1AB05	480V	—	BA-E02P	—	—	C	S	0.5	C	S	0.5	C
11566B7001M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01B	I	>30.5	C	I	>30.5	C	I	>30.5	C
11566B7003M01	DGB VENT FAN	11E	X4	1AB05	480V	50	BG-F01C	P	>30.5	C	P	>30.5	C	P	>30.5	C
11531N7001H01	CB CONTROL ROOM HTR	11E	X4	1AB05	480V	118KW	BG-C05R	N NOTE 11	>30.5	C	N NOTE 11	>30.5	C	N NOTE 11	>30.5	C
11531N7001M01	CB CONTROL ROOM FIL UNIT	11E	X4	1AB05	480V	125	BG-C01E	P	>30.5	C	P	>30.5	C	P	>30.5	C
11513H7001000	CTB HYDROGEN RECOMBINER	11E	X6	1AB05	480V	75KW	BG-B02U	M	>30.5	C	M	>30.5	C	M	>30.5	C
11805S3B15	LOAD CENTER TRANS. 1AB15	11E	X3	1AB15	4.16 -480V	1000KVA	BA-D02K	—	—	C	—	0	C	—	0	C
	SECONDARY BREAKER 1AB15	11E	X3	1AB15	480V	—	BA-E02R	—	—	C	S	0.5	C	S	0.5	C
11202W4001M01	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03A	P	>30.5	C	P	>30.5	C	P	>30.5	C
11202W4001M02	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03B	P	>30.5	C	P	>30.5	C	P	>30.5	C
11202W4001M03	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03C	P	>30.5	C	P	>30.5	C	P	>30.5	C
11202W4001M04	NUC SERV CLG TWR FAN	11E	X4	1AB15	480V	100	BD-K03D	P	>30.5	C	P	>30.5	C	P	>30.5	C
11213P6002M01	SPENT FUEL PIT PUMP	11E	X6	1AB15	480V	100	BD-H01B	M	>30.5	C	M	>30.5	C	M	>30.5	C
11561N7001M01	PIPE PEN RM FILTER/EXH FAN	11E	X4	1AB15	480V	75	BG-D01D	S	15.5	C	S	15.5	C	S NOTE 17	15.5	C
11561N7001H01	PIPE PEN FLTR/EXH UNIT HTR	11E	X4	1AB15	480V	80KW	BG-D01C	I,K NOTE 17	15.5	C	I,K NOTE 17	15.5	C	I,K NOTE 17	15.5	C
11805S3B01	LOAD CENTER TRANS. 1NB01	62E	X3	1NB01	4.16 -480V	1000KVA	BA-D02G	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	—	0	C
	SECONDARY BREAKER	62E	X3	1NB01	480V	—	BA-E01A	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	S	10.5	C
11511B7001M01	CTB CAV CLG FAN (NON 1E)	62E	X4	1NB01	480V	75	BG-B02Q	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	S	10.5	C
11805Q3PB1	FDR TO HTR PNL 1NBPB1	62E	X3	1NB01	480V	483KW	BD-B01K	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	K	10.5	C
11806B3CN5	BATTERY CHARGER 1ND3ACA	62E	X3	1NB01	480V	150KVA	BA-E03P	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	MM	10.5	C
11806B3CN1	BATTERY CHARGER 1ND1CA	62E	X3	1NB01	480V	100KVA	BA-E03F	SEE NOTE 3.a	>30.5	C	SEE NOTE 3.a	>30.5	C	MM	10.5	C
11806B3CAA	BATTERY CHARGER 1AD1CA	11E	X3	1ABA	480V	100KVA	AA-G01A	N	0	C	N	0.5	C	N	0.5	C
11806B3CCB	BATTERY CHARGER 1CD1CB (NOTE 19)	11E	X3	1ABA	480V	75KVA	AA-G01A	N	0	C	N	0.5	C	N	0.5	C
11807Y312	REG XFMR 1ABA07RX	11E	X3	1ABA	480V	7.5KVA	AA-G02A	N	0	C	N	0.5	C	N	0.5	C
11531B7002M01	CBCR CHLR RM VENT FAN	11E	X4	1ABA	480V	5.0	BG-C01B	K	0	C	K	0.5	C	K	0.5	C
A1542N7001M01	FHB PA FLTR EXH FAN	11E	X4	1ABA	480V	40	BG-E01A	K	0	C	K	0.5	C	K	0.5	C
A1542N7001H01	FHB PA EXH UNIT HTR	11E	X4	1ABA	480V	20KW	AA-F18A	I,K	0	C	I,K	0.5	C	I,K	0.5	C
11592P7001M01	CB ESF CHILLED WATER PUMP	11E	X4	1ABA	480V	30	BG-G02C	L NOTE 16A	120	C	L NOTE 16A	120	C	M	>30.5	C
11592C7001M02 &M03	CB ESF CHILLER PURGE COMP. B	11E	X4	1ABA	480V	0.75	AA-F18A	I,K NOTE 16B,C	>30.5	C	I,K NOTE 16B,C	>30.5	C	I,K NOTE 16B,C	>30.5	C
11539A7005M01	CB NORMAL AC RM ESF AC FAN MOT (NOTE 19)	11E	X4	1ABA	480V	5	BG-C07R	L	0	C	L	0.5	C	K	0.5	C
DISTR PNL 1AYA1	120/240 DISTRIBUTION PANEL	11E	X3	1ABA	480V	15KVA	AA-F18A	N	0	C	N	0.5	C	N	0.5	C
1HV-12114	MCR AIR INTAKE ISO	11E	X4	1ABA	480V	1.9	BG-C01G	K	0	10 SEC	K	0.5	10SEC	K	0.5	10SEC
1HV-12130	CBCR RTRN AIR FAN IN	11E	X4	1ABA	480V	.2	BG-C01X	I NOTE 11	>30.5	10SEC	I NOTE 11	>30.5	10 SEC	I NOTE 11	>30.5	10SEC
1HV-12118	CBCR FLTR UNIT IN	11E	X4	1ABA	480V	.2	BG-C01V	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10SEC
1HV-12128	CBCR FLTR UNIT OUT	11E	X4	1ABA	480V	.2	BG-C01X	I NOTE 11	>30.5	10 SEC	I NOTE 11	>30.5	10 SEC	I NOTE 11	>30.5	10SEC
1HV-11605	NSCW PUMP P4005 OUT	11E	X5	1ABB	480V	1.9	BD-K04Z	I	70.5	15SEC	I	70.5	15SEC	I	70.5	15SEC
1HV-8508A	AH CHG PUMP MINIFLOW	11E	X6	1ABB	480V	.7	BD-C01T	L	0	10SEC	L	0.5	10 SEC	I	0.5	10 SEC.
1HV-8509A	AH CHG PUMP MINIFLOW	11E	X6	1ABB	480V	.7	BD-C01V	M	>30.5	10SEC	M	>30.5	10 SEC	M	>30.5	10 SEC.
11807Y3IA11	INVERTER 1AD1I11	11E	X3	1ABB	480V	10KVA	AA-G01A	SEE NOTE 14	0	C	SEE NOTE 14	0.5	C	SEE NOTE 14	0.5	C
11807Y3RX22	REG XFMR 1ABB40RX	11E	X3	1ABB	480V	7.5KVA	AA-G01A	N	0	C	N	0.5	C	N	0.5	C
11202P4008M01	NS CW XFR PUMP	11E	X4	1ABB	480V	30	BD-K01B	M	>30.5	4DAYS	M	>30.5	4DAYS	M	>30.5	C
11555A7005M01	ELEC MCC ROOM COOLER	11E	X4	1ABB	480V	2.0	BG-D05E	L	0	C	L	0.5	C	K	0.5	C
11555A7011M01	CCWP ROOM COOLER FAN	11E	X4	1ABB	480V	5.0	BG-D05E	L	0	C	L	0.5	C	K	0.5	C
11555A7017M01	SFP PUMP ROOM CLR	11E	X4	1ABB	480V	5.0	BG-D05U	L	0	C	L	0.5	C	K	0.5	C
1HV-11606	NSCW PUMP P4003 OUT	11E	X5	1ABB	480V	1.9	BD-K04Z	I	70.5	15SEC	I	70.5	15SEC	I	70.5	15SEC

** SEE NOTE 21

EQUIPMENT TAG NUMBER	EQUIPMENT DESCRIPTION	PROJECT CLASSIFICATION (ELECTRICAL SYSTEM)	PROCUREMENT SPECIFICATION	BUS NUMBER	BUS VOLTAGE	NAMEPLATE RATING HP (OR KW)	REFERENCE DWG. 1X3D-	LOSS OF COOLANT ACCIDENT SI PREFERRED OFFSITE SOURCE AVAILABLE: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START (NOTE 7)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER & SUBSEQUENT SI: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.b)	RUNNING TIME AFTER START (NOTE 4)	LOSS OF OFFSITE POWER: MANUAL, PROCESS, ACDT, OR SEQUENCER START, SEE TABLE 1	TIME (SEC) TO START AFTER D-G BRKR IS CLOSED (NOTE 3.c)	RUNNING TIME AFTER START (NOTE 4)
11540B7003M01	NSCW TWR CAB TUN VENT FAN	11E	X4	1ABB	480V	3.0	BG-K01C	K	0	C	K	0.5	C	K	0.5	C
1HV-8104 *	BORIC ACID TANK TO CHARGE PUMP	11E	X6	1ABB	480V	.7	BD-C03F	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
1HV-8116 *	CHARGING PUMP 1A DISCHARGE	11E	X6	1ABB	480V	.7	BD-C05E	L	0	10 SEC	L	0.5	10 SEC	M	>30.5	10 SEC
1HV-8106	CHG PUMP TO RCS ISO	11E	X6	1ABB	480V	2.0	BD-C03H	L	0	10 SEC	L	0.5	10 SEC	M	>30.5	10 SEC
1HV-1668B	NUC SERV CL TWR A BYPASS VALVE	11E	X5	1ABB	480V	.7	BD-K05V	K	0	20 SEC MAX	K	0.5	20 SEC MAX	K	0.5	20 SEC MAX
1HV-8804A	RHR HEXCH TO CHG PUMP	11E	X6	1ABB	480V	1.6	BD-D02L	M	>30.5	15 SEC	M	>30.5	15 SEC	M	>30.5	15 SEC
1HV-8809A	RHR TNA TO SIS COLD LEG ISO	11E	X6	1ABB	480V	7.8	BD-D02V	M	>30.5	15 SEC	M	>30.5	15 SEC	M	>30.5	15 SEC
1HV-8835 *	SIS COLD LEG LOOP INLET HDR ISO	11E	X6	1ABB	480V	1.0	BD-D03H	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
1HV-1668A	NUC SERV CL TWR A ISO VALVE	11E	X5	1ABB	480V	1.6	BD-K05U	K	0	20 SEC MAX	K	0.5	20 SEC MAX	K	0.5	20 SEC MAX
1LV-0112B	VOL CONT TANK OUT ISO	11E	X6	1ABB	480V	0.7	BD-C02F	L	0	10 SEC	L	0.5	10 SEC	K	0.5	10 SEC
1PV-2550A	PIPING PEN RM A TO ATMOS	11E		1ABB	480V	.2	BG-D01L	I	15.5	10 SEC	I	15.5	10 SEC	I	15.5	10 SEC
1HV-1975 *	AUX CCW RETRN ISO	11E	X5	1ABB	480V	.66	BD-L03G	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
1HV-1979 *	AUX CCW SUPPLY ISO	11E	X5	1ABB	480V	.66	BD-L03J	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
DISTR PANEL 1AYB1	DISTRIBUTION PANEL	11E	X3	1ABB	480V	15KVA	AA-F16A	N	0	C	N	0.5	C	N	0.5	C
1PV-3000	ATMOSPHERIC DUMP VALVE	11E	X5	1ABB	480V	1.5	BC-Q03Q	K	0	20 SEC	K	0.5	20 SEC	K	0.5	20 SEC
1PV-3030	ATMOSPHERIC DUMP VALVE	11E	X5	1ABB	480V	1.5	BC-Q03Q	K	0	20 SEC	K	0.5	20 SEC	K	0.5	20 SEC
1HV-11600	NSCW PUMP P4001 OUT	11E	X5	1ABB	480V	2.0	BD-K04Z	I	70.5	15 SEC	I	70.5	15 SEC	I	70.5	15 SEC
1HV-5137	AUX FD PP P4003 DISCH MOV	11E	X5	1ABB	480V	1.3	BC-F08C	L	0	10 SEC	L	0.5	10 SEC	K	0.5	10 SEC
1HV-5139	AUX FD PP P4003 DISCH MOV	11E	X5	1ABB	480V	1.3	BC-F08D	L	0	10 SEC	L	0.5	10 SEC	K	0.5	10 SEC
11540B7005M01 *	TB AB ELEC TUNNEL VENT FAN	11E	X4	1ABB	480V	10	BG-K01E	MM	0	C	MM	0.5	C	MM	0.5	C
PNL 1AYC1	120/240 DISTRIB PANEL	11E	X3	1ABC	480V	10KVA	AA-F03A	N	0	C	N	0.5	C	N	0.5	C
11532B7001M01	CBSF BATT RM EXH FAN	11E	X4	1ABC	480V	1.5	BG-C04N	L	0	C	L	0.5	C	MM	0.5	C
11532B7003M01	CBSF BATT RM EXH FAN (STBY)	11E	X4	1ABC	480V	1.5	BG-C04Q	L	0	C	L	0.5	C	MM	0.5	C
11532A7001M01	CBSF ELEC EQUIP RM A/C	11E	X4	1ABC	480V	40	BG-C04A	L	0	C	L	0.5	C	MM	0.5	C
11539A7001M01 NOTE 19	CB AUX RELAY RM A C	11E	X4	1ABC	480V	2	BG-C07M	L	0	C	L	0.5	C	MM	0.5	C
11807Y310	REG XFMR 1ABC09RX	11E	X3	1ABC	480V	7.5KVA	AA-G01A	N	0	C	N	0.5	C	N	0.5	C
11808T3116	LTG ISOLATION XFMR 1ABC23RX	11E	X3	1ABC	480V	15KVA	AA-F03A	N	0	C	N	0.5	C	N	0.5	C
11807Y3RX25	ROD POS IND CAB ISO XFMR 1ABC20RX	11E	X3	1ABC	480V	7.5KVA	AA-G03B	N	0	C	N	0.5	C	N	0.5	C
1HV-2584A	CTB COOLING UNIT (DAMPER)	11E	X4	1ABC	480V	.13	BG-B03X	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
1HV-2584B	CTB COOLING UNIT (DAMPER)	11E	X4	1ABC	480V	.13	BG-B03X	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC
1HV-12742	CBSF BATTERY RM EXH	11E	X4	1ABC	480V	.2	BG-C04S	I	0	10 SEC	I	0.5	10 SEC	I	0.5	10 SEC
1HV-2628A	CTB NORM PURGE EXH ISO	11E	X5	1ABC	480V	1.33	BG-B04G	L	0	10 SEC	L	0.5	10 SEC	M	>30.5	10 SEC
1HV-12748	CBSF BATTERY RM FAN (STBY)	11E	X4	1ABC	480V	.2	BG-C04U	I	0	10 SEC	I	0.5	10 SEC	I	0.5	10 SEC
1HV-8808C	ACCUM LOOP NO. 3	11E	X6	1ABC	480V	10.3	BD-D02T	L	0	10 SEC	L	0.5	10 SEC	K	0.5	10 SEC
1HV-19055 *	THERM BARR CW RCP 003	11E	X4	1ABC	480V	.7	BD-L03M	K	0	10 SEC	K	0.5	10 SEC	K	0.5	10 SEC
1HV-19057 *	THERM BARR CW RCP 004	11E	X4	1ABC	480V	.7	BD-L03N	K	0	10 SEC	K	0.5	10 SEC	K	0.5	10 SEC
1HV-8146	CHARG TO REAC COOL SYS ISO	11E	X6	1ABC	480V	.7	BD-C03T	M	>30.5	10 SEC	M	>30.5	10 SEC	M	>30.5	10 SEC

NOTES:

1.) LOSS OF OFFSITE POWER (LOP) IS DEFINED AS A LOSS OF VOLTAGE ON THE 4.16KV BUS.

2.)
a. TIME t=0 SECONDS IS DEFINED AS THE TIME THE DIESEL GENERATOR BREAKER IS CLOSED FOR ALL LOP CONDITIONS AND THE TIME THE SI SIGNAL IS RECEIVED IN THE SAFETY FEATURES (SF) SEQUENCER WITH PREFERRED OFFSITE SOURCE AVAILABLE TO THE 4.16KV BUS.
b. DURING LOP CONDITIONS THE MAXIMUM TIME AFTER THE D.G. RECEIVES THE START SIGNAL TO THE FIRST LOAD STEP IS 10 SECONDS. THIS 10 SECOND INTERVAL INCLUDES A MAXIMUM OF 9.5 SECONDS FOR THE DIESEL GENERATOR TO START AND COME UP TO RATED VOLTAGE AND FREQUENCY PLUS 0.5 SECONDS, AFTER THE DIESEL GENERATOR BREAKER CLOSES, TO ENERGIZE THE UNLOADED 480V LOADCENTER TRANSFORMERS.

3.)
a. 15 THE 4.16KV BREAKERS SUPPLYING 1NB01 AND 1NB10 15 WILL BE TRIPPED DURING SI CONDITIONS BY SSPS. POWER CAN BE RESTORED MANUALLY TO BUS 1NB01 AND 1NB10 UNDER ADMINISTRATIVE PROCEDURE AFTER THE SI SIGNAL HAS BEEN MANUALLY BYPASSED.
b. DURING LOP CONDITIONS AND SUBSEQUENT SI ALL BUSSES (ECEPT 1NB01, 1NB10, 1NBS, 1NBR, 1NBO AND 1NBI) WILL HAVE POWER AVALIABLE FOR LOADING 0.5 SECONDS AFTER THE DIESEL GENERATOR BREAKER CLOSES.
c. DURING LOP CONDITIONS ALL BUSSES WILL HAVE POWER AVAILABLE FOR LOADING 0.5 SECONDS AFTER THE DIESEL GENERATOR BREAKER CLOSES.

4.)
a. C MEANS CONTINOUS S MEANS STANDBY
b. UNLESS OTHERWISE INDICATED MOV(S) ARE ASSUMED TO OPERATE FOR 10 SECONDS ONLY.

5.) THIS PUMP WILL BE SEQUENCED ON ONLY IF ONE OR BOTH OF THE PRIMARY PUMPS FAIL TO START ON THE PREVIOUSLY SEQUENCED STEP

6.) THE SF SEQUENCER WILL ONLY PERMIT THIS LOAD TO START FOR 1 SECOND PERIODS OF TIME AT THE BEGINNING OF THE INDICATED SEQUENCE STEP, ALL SUBSEQUENT STEPS AND AT ANYTIME AFTER 30.5 SECONDS

7.) THE FIRST LOAD STEP WILL OCCUR (WITH PREFERRED OFFSITE SOURCE AVAILABLE) 0.5 SECONDS AFTER THE SI SIGNAL IS RECEIVED AT THE SF SEQUENCER

10.) UNDER SI CONDITIONS, THESE LOADS ARE SEQUENCED ON AT LOW SPEED ONLY FOR LOP CONDITIONS THEY ARE SEQUENCED ON AT HIGH SPEED ONLY

11.) THESE LOADS ARE INTERLOCKED WITH OTHER LOADS WHICH ARE BLOCKED BY SEQUENCER UNITL 30.5 SECONDS

12.) TRAIN B SIMILAR EXCEPT WHERE NOTED BY ASTERISK (*) OR IN TRAIN B TABLE (SEE NOTES 18,19 &20)

13.) SI MEANS SAFETY INJECTION SYSTEM SIGNAL ACTUATED.

14.) THIS LOAD FED FROM BREAKER CLOSED ONLY DURING STARTUP OF THE LOAD

15.) SEQUENCER PROVIDES MOTOR AUTO-START CIRCUIT WITH START SIGNAL AT 30.5 SECOND STEP. HOWEVER MOTOR WILL START AT 50.5 SECONDS DUE TO AN ADDITIONAL TIME DELAY OF 20 SECONDS BY AGASTAT TIME DELAY RELAY IN THE AUTO-START CIRCUIT.

NOTES CONT

16 A) SI PROVIDES CB ESF CHILLER WATER PUMP AUTO-START CIRCUIT WITH START SIGNAL. HOWEVER PUMP MOTOR START WILL BE DELAYED APPROXIMATELY 120 SEC. DUE TO ADDITION OF AGASTAT TIME DELAY RELAY IN THE AUTO-START CIRCUIT.

B) CB ESF CHILLER PURGE COMP MTR IS INTERLOCKED WITH THE ESF CHILLER COMPRESSOR WHICH IS BLOCKED BY THE SEQUENCER UNTIL 30.5 SEC

C) CB ESF CHILLER OIL PUMP MOTOR IS INTERLOCKED WITH ESF CHILLER CHILLED WATER FLOW SWITCH. THIS SWITCH OPERATES DUE TO FLOW FROM THE CB ESF CHILLED WATER PUMP APPROXIMATELY 120 SECONDS AFTER THE SI SIGNAL IS INITIATED.

17. FAN 1-1561-N7-001 WILL LOAD SHED BUT WILL NOT AUTO-START UNLESS A CVI SIGNAL IS PRESENT. HEATER 1-1561-N7-001-H01 IS INTERLOCKED WITH FAN 1-1561-N7-001 AND IS ALSO PROCESS CONTROLLED.

TABLE 1

INITIATING LETTER DESIGNATOR	TIME SEC TO OPERATE AFTER DIESEL GENERATOR BREAKER CLOSES	STARTING INITIATED BY
K	0.5	PROCESS DEVICES (THERMOSTATS PRESSURE SWITCHES ETC)
L	0.5	ACCIDENT SIGNAL (SI) OR LOP
M	>30.5	MANUAL (ASSUME NO MANUAL ACTIONS WILL OCCUR UNTIL SEQUENCER TIMING CYCLE IS COMPLETED)
N	0.5	PANELS & OTHER LOADS FED FROM BRKRS ONLY-WHICH REMAIN CLOSED
P	>30.5 (BLOCKED BY SEQUENCER)	PROCESS DEVICE
S	<30.5	AUTOMATICALLY BY SEQUENCER CAN BE STOPPED MANUALLY ONLY AFTER 30.5 SEC
MM	0.5	MNL STARTED WITH MEMORY (SLIP OR MNTN START CONT) LOAD WILL RE-START UPON RESTORATION OF POWER
I	VARIOUS	INTERLOCKED WITH OTHER LOADS

NOTES CONT.

18. THOSE LOADS MARKED BY AN ASTERISK (*) ARE UNIQUE TO THEIR PARTICULAR TRAIN AND DO NOT HAVE AN EQUIVALENT OPPOSITE TRAIN LOAD.

19. THESE LOADS DIFFER FROM THEIR CORRESPONDING OPPOSITE TRAIN LOAD. BOTH LOADS ARE LISTED UNDER THEIR RESPECTIVE TRAIN.

20. TRAIN A AND B LOADS WHICH DIFFER BY .2 HP OR LESS ARE LISTED ONLY UNDER TRAIN A. ALL OTHER LISTED DATA IS SIMILIAR.

21. THE UNIT 1, TRAIN A CENTRIFUGAL CHARGING PUMP MOTOR (11208P6002M01) WILL PRODUCE 715 HORSE-POWER DURING LOCA CONDITIONS.

NUCLEAR SAFETY RELATED A B

Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

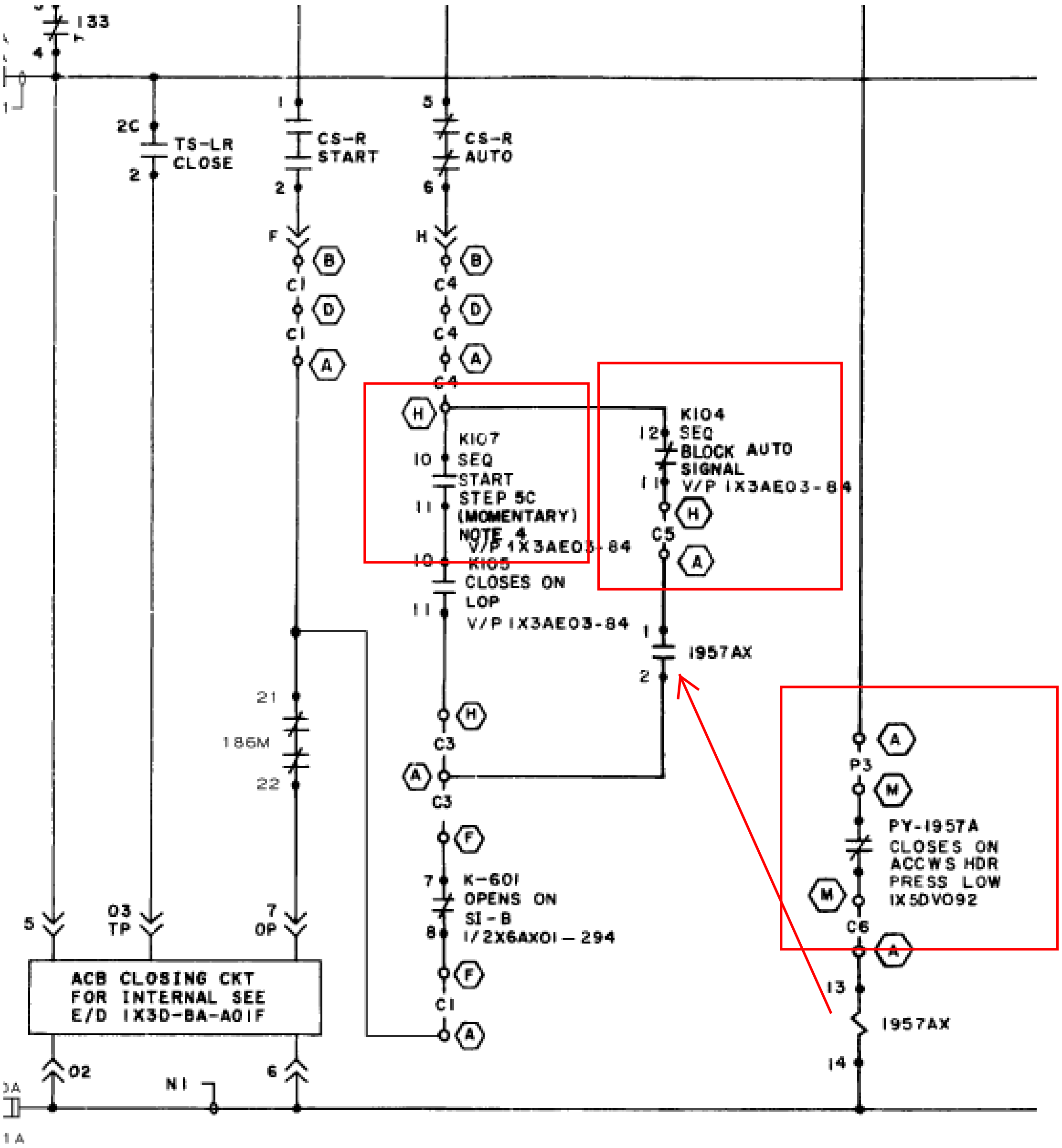
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

DIESEL GENERATOR
1A AND TRAIN "A" AC BUSSES
LOADING TABLE - PAGE 1 OF 2

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
15	INCORP. PER ABN-36788	11-28-01	CD	EOG	MWD	

SCALE: NONE	DRAWING NO.	REV.
JOB NO. 10604	1X3D-AA-K02A	15

SIZE E 34X44



133
TS-LR
CLOSE
CS-R
START
CS-R
AUTO
K107
SEQ
START
STEP 5C
(MOMENTARY)
NOTE 4
V/P 1X3AE03-84
K104
SEQ
BLOCK AUTO
SIGNAL
V/P IX3AE03-84
K105
CLOSES ON
LOP
V/P IX3AE03-84
1957AX
186M
K-601
OPENS ON
SI-B
1/2X6AX01-294
PY-1957A
CLOSES ON
ACCWS HDR
PRESS LOW
IX5DV092
1957AX
ACB CLOSING CKT
FOR INTERNAL SEE
E/D IX3D-BA-A01F

1. 077AK3.02 001/LOCT/RO/M/F 3.6/3.9/077AK3.02/LO-TA-60043///

Initial condition:

- Unit 1 is at 100% reactor power.

Current condition:

- 18017-C, "Abnormal Grid Disturbances / Loss of Grid," Section 'A' for "Degraded Grid Conditions," is in progress.

Which one of the following completes the following statement?

The DGs are maintained in standby alignment to __(1)__,

and

the Main Generator is operated within the acceptable region of the reactive capability curve to prevent damage to the __(2)__ due to overheating.

A.✓ (1) prevent a grid disturbance from impacting availability

(2) Main Generator

B. (1) prevent a grid disturbance from impacting availability

(2) reactive loads

C. (1) comply with the required Tech Spec alignment for operability

(2) Main Generator

D. (1) comply with the required Tech Spec alignment for operability

(2) reactive loads

K/A

077 **Generator Voltage and Electric Grid Disturbances**

AK3.02 **Knowledge of the reasons for the following responses as they apply to the Generator Voltage and Electric Grid Disturbances:**

- Actions contained in abnormal operating procedure for voltage and grid disturbances.

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of the reasons for placing the EDG's in standby alignment as directed by the grid distrubance AOP and the reason for ensuring

the main generator is operated within the capacity curve during a grid disturbance.

EXPLANATION OF REQUIRED KNOWLEDGE

Per AOP 18017-C step A1, DG's are checked "IN STANDBY". The RNO to this step is to "restore DG's to operable status". The RNO is somewhat misleading in that a DG is OPERABLE when paralleled to the Grid. AOP 18017-C does not have a background document, however the intent of this step is to place the DG's in a standby alignment, not just for them to be OPERABLE. Lesson plan V-LO-LP-60330 states "it is not desirable to go ahead and start the engines and run unloaded or even parallel to the bus". The lesson plan cites issues with cylinder loading. Additionally, IN 84-69 "SUPPLEMENT 1: OPERATION OF EMERGENCY DIESEL GENERATORS," states that EDGs should not be paralleled to the Grid during times of instability to prevent the Grid from damaging the EDG and rendering both AC sources to the bus incapable of supplying emergency power. The safest configuration for the EDGs is in standby.

Per AOP 18017-C step A5, the Main Generator is to be maintained within the capability curve of Figure 1. Figure 1 depicts the generator operational limits associated with the bounding curves. Additonally, lesson plan V-LO-LP-60330 states that if the main generator cannot be maintained within the capability curve, the reactor and generator are tripped to prevent damage to the generator windings due to frequency/voltage swings.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. The reason for maintaining the diesel generators in standby alignment is to eliminate or reduce the potential for a grid disturbance impacting both the normal and emergency sources of power at the same time.

The second part is correct. The reason for maintaining the main generator operations within the prescribed region of the reactive curve is to prevent overheating of the generator components themselves to include the stator winding, rotor, and conductors.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. The candidate must consider the potential damage to components due to excessive VAR loading output by the main generator. Being outside the Tech Spec required voltage (either above or below limits) can damage components. In a degraded grid condition, however, the main generator would actually respond to attempt to maintain the grid voltage within limits, not drive voltage outside the limits. This automatic control action exposes the main generator to potential damage and the stem focuses on the main generator, not the impacted loads.

C. Incorrect. Plausible. The first part is incorrect. The reason for maintaining the diesel

generators in standby alignment is to eliminate or reduce the potential for a grid disturbance impacting both the normal and emergency sources of power at the same time. However, the candidate must consider the Tech Spec (TS) required alignment. Tech Specs refer to the diesel generator as the standby source which implies a required alignment. The candidate may not remember that the diesel generator is considered OPERABLE either in standby or parallel condition and assume the EDG is only OPERABLE in standby. 18017-C step A1 tends to back up this false logic in that the step states "Check... in standby" and the RNO states, "restore... to operable status."

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G1	
K/A#	077AK3.02	
Importance Rating:	3.6 / 3.9	
Technical Reference:	18017-C, Rev 9.5, pages 3 & 7 Lesson Plan V-LO-LP-60330, Rev 4.0, pages 7 & 8 INPO IN 84-69, Supplement 1, February 24, 1986	
References provided:	None	
Learning Objective:	LO-LP-60330-03	State the reasons for tripping the reactor in AOP-18017-C.
	LO-LP-60330-06	State the control room indications for a loss of grid.
	LO-LP-60330-07	State the consequences of operating the 1E 4160 buses outside the voltage limits listed in AOP-18017.
	LO-LP-60330-09	Given a set of control room indications/notifications, determine if entry conditions for either section of 18017 is met.
	LO-LP-60330-10	Given that a loss of grid has occurred from 100% power, describe the expected flowpath through the EOPs and AOP 18017-C.
	LO-LP-60104-05	State why, when anticipating an LOP, the Control Room Operator is instructed NOT to start the diesel generator(s).
	LO-TA-60043	Respond to an Abnormal Grid Disturbance per 18017-C
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.4 / 41.5/ 41.7/ 41.10 / 45.8	
Comments:		

You have completed the test!

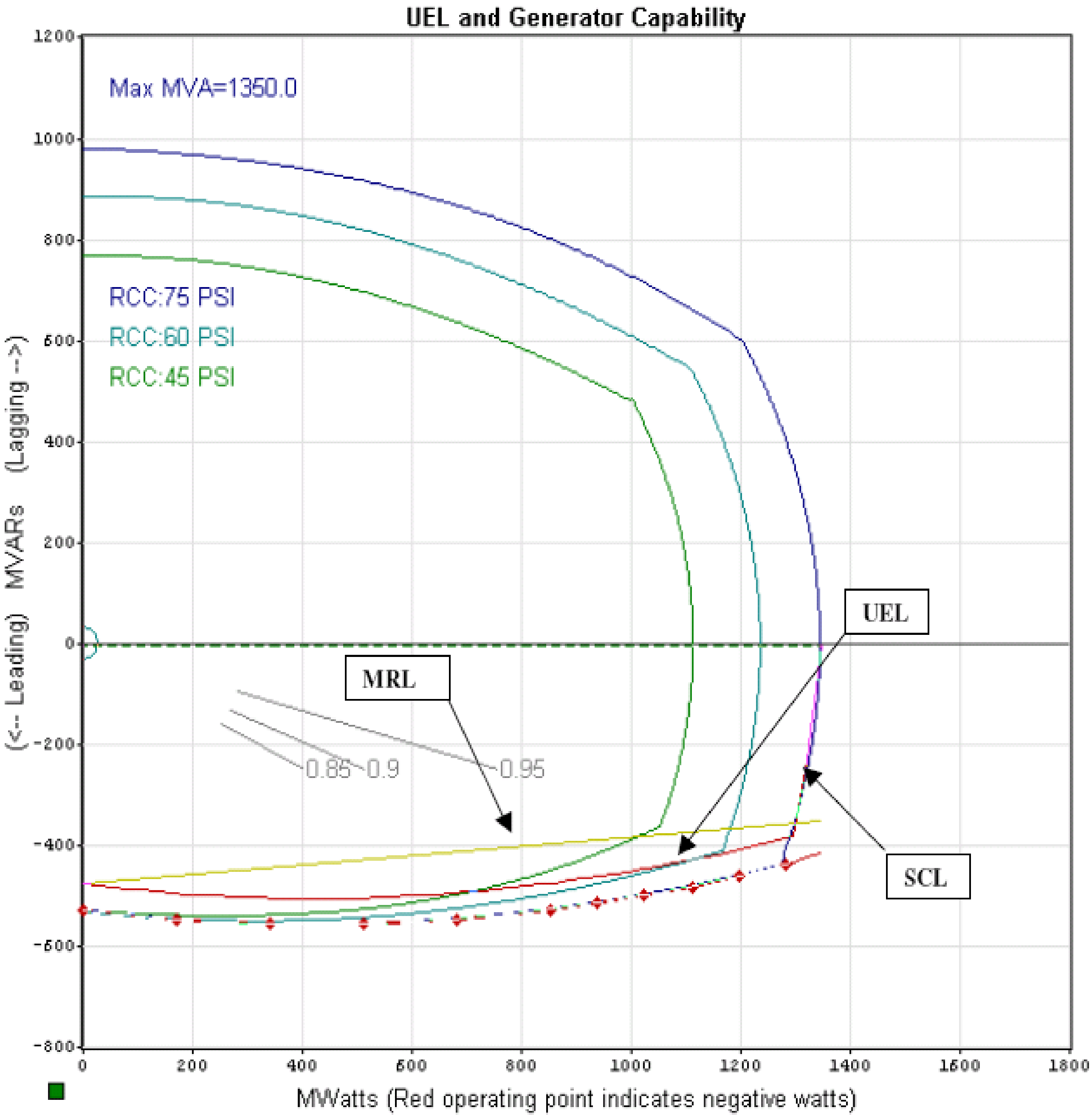
A. DEGRADED GRID CONDITIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A1. Check Diesel Generators - IN STANDBY.	A1. Restore Diesel Generators to operable status.
A2. Terminate maintenance or testing activities on critical electrical distribution components.	
A3. Check Main Generator Power System Stabilizer on COI - PSS ENABLED.	A3. Perform actions of TABLE 1, as necessary.
A4. Initiate the Continuous Actions Page.	
*A5. **Maintain Main Generator - OPERATING WITHIN THE REACTIVE CAPABILITY CURVE OF FIGURE 1.**	A5. IF Main Generator can NOT be maintained within the capability curve, THEN trip the reactor and initiate 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION.
A6. Place the following on alternate power supply using 13800, MAIN TURBINE OPERATION: • Main Turbine Turning Gear • Turning Gear Oil Pump	
A7. Verify Turning Gear Oil Pump: With turbine on line - IN AUTO. -OR- With turbine on turning gear - IN OPERATION.	Verify Main Turbine Auxiliary Emergency DC Lube Oil Pump is operating.

FIGURE 1

Sheet 1 of 1

REACTIVE CAPABILITY CURVE



2) A DG paralleled to the grid during degraded voltage conditions may trip if conditions worsen just when it will be called upon to start. Local operator actions may be necessary to return the DG to service. The DG could also sustain damage due to the transient.

13145

Precaution 2.1.10

The SOPs 13145 have the following precaution:

The Diesel Generators should NOT be operated in parallel with the offsite grid for prolonged periods of time. This is to keep disturbances in the grid from affecting the Diesel Generators.

The OSPs 14980 have the following precautions:

During surveillance testing, only one DG shall be paralleled at a time to the off site power source.

14980

Precautions 3.11

and 3.18

To serve as a dependable backup power source, a DG should be kept separate from the offsite source if it is the only OPERABLE diesel. The DG should remain in standby and only be paralleled with an offsite source to meet surveillance requirements. Parallel operations may be conducted as a part of a preplanned activity if a supporting risk assessment has been completed.

3. Any maintenance in progress on critical electrical equipment should be stopped.

   a This equipment should be restored to service if possible.

   b If the maintenance is being performed due to equipment inoperability, the SS should evaluate the need to continue work in an attempt to restore the component to service.

4. Make sure Main Generator Power System Stabilizer in service

   Review Table 1

   a. PSS increases the EX2100 AC regulator control stability by negating the regulators inherent lag response to a problem which left to itself tends to destabilize generator output during relatively small transients.

   b. The action of the PSS is to increase stability and prevent the voltage Regulator from driving the system into increasing amplitude oscillations.

5. Maintain proper Main Generator operation within the “Reactive Capability Curve” of 13830, Main Generator Operation.

   Objective 3
   Review Capability Curve Useage

   a. If it cannot be maintained within the capability curve, the Reactor Is required to be tripped to prevent damage to generator windings due to frequency/voltage swings and Emergency Procedures entered.

   Objective 3, 4, 5, 8

6. Inform shift personnel of the grid condition and stress the potential for a loss of site power.

   a. The intent is to ensure everyone has a chance to prepare and implement contingency actions should the loss of power occur.

   b. These actions may include the call in of additional personnel or at least placing them on call to more readily respond if needed.

2. Notification from the Power Control Center that the distribution center is "one contingency away" from being unable to maintain system voltage between 230 and 242 kV.

   Entry into Section A

   a. The statement that the distribution center is "one contingency away" from being able to maintain grid voltage within limits means that one more system fault will cause a degraded system to exist.

3. Annunciator Response Procedures 17036 and 17037, SEQ TROUBLE windows, if a degraded grid condition exists.

   Entry into Section A

4. ***Section "B"*** is entered when All Offsite Power are Deenergized

   Objective 9

C. PROCEDURE USAGE-SECTION "A" DEGRADED GRID VOLTAGE

1. Purpose of Section A - The intent is to ensure the plants vital equipment is available for a Loss of Offsite Power.

2. The Diesel Generators are verified to be operable and in standby.

   a. The DGs are the backup emergency power supplies for the 1E bus.

   b. It is important to note that it may be desirable to station someone at the DG to ensure it starts and runs properly in the event a degraded voltage condition eventually develops.

   c. It is not desirable to go ahead and start the engines and run unloaded or even parallel to the bus.

      SMART – Ensuring DG availability

      1) Extended unloaded or low load operation results in the buildup of combustion products in the engine exhausts due to incomplete fuel burn in the cylinders.

         a) These products can eventually cause fouling of cylinder valves

         b) preventing proper operation and reduction in the load carrying capacity of the DG.

III. LESSON OUTLINE

2) A DG paralleled to the grid during degraded voltage conditions may trip if conditions worsen just when it will be called upon to start. Local operator actions may be necessary to return the DG to service. The DG could also sustain damage due to the transient.

13145

Precaution 2.1.10

The SOPs 13145 have the following precaution:

The Diesel Generators should NOT be operated in parallel with the offsite grid for prolonged periods of time. This is to keep disturbances in the grid from affecting the Diesel Generators.

The OSPs 14980 have the following precautions:

During surveillance testing, only one DG shall be paralleled at a time to the off site power source.

14980

To serve as a dependable backup power source, a DG should be kept separate from the offsite source if it is the only OPERABLE diesel. The DG should remain in standby and only be paralleled with an offsite source to meet surveillance requirements. Parallel operations may be conducted as a part of a preplanned activity if a supporting risk assessment has been completed.

Precautions 3.11

and 3.18

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   a This equipment should be restored to service if possible.

   b If the maintenance is being performed due to equipment inoperability, the SS should evaluate the need to continue work in an attempt to restore the component to service.

4. Make sure Main Generator Power System Stabilizer in service

Review Table 1

   a. PSS increases the EX2100 AC regulator control stability by negating the regulators inherent lag response to a problem which left to itself tends to destabilize generator output during relatively small transients.

   b. The action of the PSS is to increase stability and prevent the voltage Regulator from driving the system into increasing amplitude oscillations.

5. Maintain proper Main Generator operation within the “Reactive Capability Curve” of 13830, Main Generator Operation.

Objective 3
Review Capability Curve Useage
Objective 3, 4, 5, 8

   a. If it cannot be maintained within the capability curve, the Reactor Is required to be tripped to prevent damage to generator windings due to frequency/voltage swings and Emergency Procedures entered.

6. Inform shift personnel of the grid condition and stress the potential for a loss of site power.

   a. The intent is to ensure everyone has a chance to prepare and implement contingency actions should the loss of power occur.

   b. These actions may include the call in of additional personnel or at least placing them on call to more readily respond if needed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

February 24, 1986

Information Notice No. 84-69, SUPPLEMENT 1: OPERATION OF EMERGENCY DIESEL GENERATORS

Addresses:

All nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

Information Notice 84-69, issued on August 29, 1984, was provided to alert recipients of potentially significant safety problems that can arise when one or more emergency diesel generators (EDGs) are operated in modes other than the prescribed standby service mode, such as loaded on non-emergency buses parallel with offsite power sources. The purpose of this supplement is to reemphasize the need for licensees to review the information provided in IN 84-69, in addition to the information contained herein, for applicability to their facilities and consider actions, if appropriate, to preclude similar problems at their facilities. However, suggestions contained in this supplement do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Following a 10 CFR 50.72 report made to the NRC Headquarters Operations Center on August 12, 1985, it was discovered that Crystal River Unit 3 was continuously running the one operable EDG loaded in parallel with the grid while the other EDG was declared inoperable. Crystal River Technical Specifications require fast starting of the operable EDG (i.e., verifying that the diesel starts from ambient conditions and accelerates to the required speed within a required period of time) within 1 hour after the declaration of an inoperable EDG and every 8 hours thereafter. Because of a concern about increased EDG wear and reduced overall EDG reliability, the licensee chose to keep the EDG running loaded parallel to the offsite grid rather than fast starting the EDG every 8 hours.

The licensee believed that continuous running was an acceptable alternative to the test starts required by the Technical Specifications and that the EDG was operable per Technical Specifications while running in parallel with the offsite power system. The licensee indicated also that it was aware of IN 84-69 and had implemented procedures that prohibited operating the EDG parallel to the grid during inclement weather (e.g. , lightning, heavy

winds).

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Discussion:

When an EDG is operated connected to offsite or nonvital loads, the emergency power system is not independent of disturbances on the nonvital and offsite power systems that can adversely affect emergency power availability. The situation is of particular concern when the onsite emergency power system is already in a degraded condition due to an EDG being inoperable and the operable EDG is loaded on non-emergency loads. In this condition, a disturbance in the non-emergency power system could result in both a loss of offsite power and a disabling of the remaining emergency power source. Although the events described in IN 84-69 occurred due to weather conditions, the concerns of the IN apply to parallel operation of EDGs with non-emergency loads at all times.

If a fault develops while the EDG is connected to non-emergency buses, EDG availability for subsequent emergency demands may be affected. In some design configurations, the EDG would trip as a result of overcurrent or reverse power, actuate a lockout device, and require local operator action to reset the lockout. In such cases, the EDG is recoverable, but the timeliness of its availability is not comparable to that of having the EDG in its normal standby service.

In other design configurations the EDG may not trip, but the operation of the load sequencer may be adversely affected. The load sequencer timers are often linked with the closing of the EDG output breaker or with detection of loss of voltage on the bus. If the EDG does not trip, conditions are not proper for the designed operation of the load sequencers. Consequently, the EDG cannot perform automatically in a manner comparable to that of having the EDG in its normal standby mode.

Another potential concern deals with the vulnerability of the EDG to trip signals which are bypassed for emergency demands but are operable for manual starts and during running for test purposes. The EDG would be more vulnerable to such trips.

The licensee's concern regarding excessive test starts is valid. In this particular case, the licensee was encouraged to address that concern more directly by submitting changes to the plant Technical Specifications. Such changes were approved for North Anna Unit 2 on April 25, 1985.

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No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contacts: Joseph G. Giitter, IE
(301) 492-9001

J. T. Beard, NRR
(301) 492-7465

Attachment: List of Recently Issued Information Notices

1. 078A4.01 001/LOIT AND LOCT/RO/C/A 3.1/3.1/078A4.01/LO-TA-02013///079A4.01

Initial conditions:

- Unit 1 is at 100% reactor power.
- Air Compressors #1 and #3 are in service.

Current conditions:

- ALB01-C06 SERVICE AIR HDR LO PRESS is received.
- 1PI-9377, Service Air Header Pressure, on the QMCB lowered to 92 psig and is now stable at 105 psig.

- NO operator action has been taken.

Which one of the following completes the following statement?

All air compressors in AUTO-PTL __(1)__ currently running,

and

1-PV-9375, Service Air Dryer Inlet Isolation Valve, __(2)__ closed.

	__(1)__	__(2)__
A.	are	is
B.✓	are	is NOT
C.	are NOT	is
D.	are NOT	is NOT

K/A

078 Instrument Air

A4.01 Ability to manually operate and/or monitor in the control room:

- Pressure gauges.

K/A MATCH ANALYSIS

The question matches the KA by testing the candidate's abillity to determine the expected system response as the instrument air system pressure indication indication lowers in the main control room. The candidate is required to monitor air pressure and determine how the instrument air system will respond.

EXPLANATION OF REQUIRED KNOWLEDGE

AOP 18028-C requires the operator to verify that numerous automatic actions occur on lowering Instrument Air header pressure. Per SOP 13710-1 (Note prior to section 4.2.1) and 17001-C ALB01-C06, all air compressors with handswitches in AUTO PTL will start at a pressure of 100 psig as sensed at its local controller. This is a silent action because no control room annunciator alerts the operator that the setpoint has been reached. All air compressors that auto start continue to run until manually stopped by the operator.

Per ARP 17001-1, ALB01-C06 SERVICE AIR HDR LO PRESS will alarm at 95 psig. There are no equipment actuations associated with this setpoint. If pressure continues to lower, service air isolation valve PV-9375 will close at a header pressure of 80 psig. PV-9375 requires two conditions to reopen: First, PI-19380 must read >97 psig. Second, PSL-9375 must be reset locally.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The standby air compressors automatically start at 100 psig on lowering header pressure, as sensed at its local controller.

The second part is incorrect. The alarm setpoint for ALB01-C06 is 95 psig. No equipment actuations occur at this setpoint - it provides an alarm only. Service air isolation valve PV-9375 will close at a header pressure of 80 psig. The lowest pressure observed was 92 psig. Therefore, PV-9375 is open. However, candidates often confuse the annunciators associated with lowering instrument air pressure and it is common for candidates to encounter ALB01-C06 and believe that it alerts the operator to the closure of PV-9375.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. The alarm setpoint for ALB01-C06 is 95 psig. No equipment actuation occur at this setpoint - it provides an alarm only. Service air isolation valve PV-9375 will close at a header pressure of 80 psig. The lowest pressure observed was 92 psig. Therefore, PV-9375 is open.

C. Incorrect. Plausible. The first part is incorrect. The standby air compressors automatically start at 100 psig on lowering header pressure, as sensed at its local controller. However, it is a common misconception that air compressors will auto stop if header pressure rises above 100 psig. This was true for the original air compressor controllers, but is not part of the current control system. In addition, other components like the RMWST and boric acid pumps do auto stop when there is no longer a demand signal present.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO
Tier # / Group #	T2 / G1
K/A#	078A4.01
Importance Rating:	3.1 / 3.1

Technical Reference: ARP 17001-1, Rev 31.1, pages 23, 33, & 34
SOP 13710-1, Rev 39.0, page 15

References provided: None

Learning Objective:

LO-LP-60321-06	Describe the operator actions required during normal full power operation when instrument air header pressure fails below 80 psig and/or below 70 psig.
LO-PP-02101-09	List the sequence of major events on a decreasing instrument air pressure condition.
LO-TA-02013	Respond to a loss of instrument air using 18028-C

Question origin: BANK

Cognitive Level: C/A

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments:

You have completed the test!

Question Number: **53**

K/A: 078 A4.01
Ability to manually operate and/or monitor in the control room: Pressure gauges

Tier:	**2**	**RO Imp:**		**RO Exam:**	**53**	**Cognitive Level:**	**Low**
Group:	**1**	**SRO Imp:**	**n/a**	**SRO Exam:**	**53**	**Source:**	**WBN Bank**

Applicable 10CFR55 Section: (CFR: 41.7 / 45.5 to 45.8)

Learning Objective: 3-OT-SYS032A, Obj. 16, List the events and their corresponding set points that take place on decreasing control air pressure.

References: AOI-10, "Loss of Control Air," Rev. 38

Question: 53

Which ONE of the following completes the sentence below for a lowering control air system pressure?

The setpoint at which the Auxiliary Air Compressors start is __(1)__ psig. If pressure continues to lower to ___(2)___ psig, air to the Reactor Building will automatically isolate.

	(1)	(2)
a.	83	75
b.	83	70
c.	79.5	75
d.	79.5	70

DISTRACTOR ANALYSIS

a. Incorrect. Plausible since the Auxiliary Air Compressor starts at 83 psig, but the essential and non-essential air systems do not isolate at 75 psig. 75 psig is plausible since this is the required pressure for reopening air to containment valve.
b. CORRECT. Auxiliary Air Compressor starts at `83 psig, and the essential and non-essential air systems isolate at 70 psig.
c. Incorrect. Plausible since the auxiliary air system isolates at 79.5 psig, but the essential and non-essential air systems do not isolate at 75 psig. 75 psig is plausible since this is the required pressure for reopening air to containment valve.
d. Incorrect. Plausible since the auxiliary air system isolates at 79.5 psig, but the essential and non-essential air systems isolate at 70 psig.

INITIALS

4.2 SYSTEM OPERATION

NOTE

During normal operation, one or more compressors should be operated in manual with their respective handswitches in Normal After Start. The remaining available compressor(s) should be in Standby with their handswitch(es) in AUTO PTL. If the system load increases such that the base load compressor(s) cannot supply the demand, the Standby compressor(s) starts at 100 psig as sensed at its local controller. □

4.2.1 Placing An Air Compressor In Standby

4.2.1.1 **Verify** the following for the compressor to be placed in standby:

- TPCCW (TPCCW or Utility Water for rotary only) is available to the compressor. ________
- Compressor lube oil level is normal. ________

CAUTION

To allow the residual voltage restart relays to reset, the compressor should be stopped per Section 4.3.1 or 4.3.2 and allowed to remain in STOP for at least three seconds before placing the handswitch in AUTO PTL position. □

4.2.1.2 IF the air compressor is running, **shut down** compressor per Section 4.3. ________

4.2.1.3 On QMCB, **place** the desired compressor handswitch(es) in AUTO PTL position:

COMPRESSOR	HANDSWITCH	
1-2401-C4-501	1HS-19338	________
1-2401-C4-502	1HS-9383	________
1-2401-C4-503	1HS-9382	________
A-2401-C4-504	1HS-9381	________

WINDOW B06

ORIGIN	SETPOINT	
1-PSL-19414	70 psig	INSTR AIR EQUIP LO PRESS

1.0 **PROBABLE CAUSE**

1. Instrument Air Dryer, Prefilter or Afterfilter clogged.
2. System piping leak.
3. System valve misalignment.
4. Loss of all Air Compressors.

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

Go To 18028-C, "Loss Of Instrument Air."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORYOPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES:1X3D-BH-R50L, 1X4DB175-2, CX5DT1101-95B

WINDOW C06

ORIGIN	SETPOINT	
1-PSL-9375	95 psig	SERVICE AIR HDR LO PRESS

1.0 **PROBABLE CAUSE**

1. Excessive service air demand.
2. Air Compressor trip.
3. System leak.
4. Standby compressor failed to start.

2.0 **AUTOMATIC ACTIONS**

1. Service Air Dryer Inlet Isolation Valve 1-PV-9375 closes at a service air pressure of 80 psig.
2. Any standby air compressor with its handswitch in AUTO-PTL position will auto start at a discharge pressure of 100 psig decreasing.

3.0 **INITIAL OPERATOR ACTIONS**

NONE

WINDOW C06
(Continued)

4.0 **SUBSEQUENT OPERATOR ACTIONS**

1. **Check** QMCB indications and **start** a standby Air Compressor if necessary to maintain service air header pressure above 100 psig.

2. **Dispatch** an operator to check for system leaks or excessive air usage.

3. IF pressure continues to fall and CANNOT be restored, **refer to** 18028-C, "Loss Of Instrument Air".

4. Refer to 13710-1, "Service Air System" and **verify** Air Compressors are operating properly.

5. IF equipment failure is indicated, **initiate** maintenance as required.

CAUTION

Procedure 13710-1 “Service Air System” should be referenced prior to performing the following step if service air has isolated due to low pressure.

6. WHEN service air header pressure is greater than 97 psig as read on 1-PI-19380 on panel PMEC, **reset** 1-PSL-9375. Switch is located on instrument rack 15 (1-1624-P5-R15) on Turbine Building level 1 near Powdex vessels.

5.0 **COMPENSATORYOPERATOR ACTIONS**

1. **Trend** the Service Air System pressure on the Plant Computer.

2. **Check** the Service Air System pressure greater than or equal to 100 psig once per hour, and **initiate** the appropriate Subsequent Operator Actions IF pressure is low.

3. **Log** corrective actions to repair the disabled annunciator or reasons for no action on 10018-C, "Annunciator Control", Figure 2.

4. **Log** compensatory actions on 10018-C, "Annunciator Control", Figure 5.

END OF SUB-PROCEDURE

REFERENCES:1X3D-BH-R50L, 1X4DB175-2, CX5DT1101-95A

1. 086K4.02 001/LOIT/RO/M/F 3.0/3.4/086K4.02/LO-TA-43003///

Initial condition:

- Unit 1 at 100% reactor power.

Current condition:

- The running fire water jockey pump trips.

Which one of the following completes the following statement?

The standby fire water jockey pump __(1)__ automatically start as fire header pressure lowers,

and

the electric fire pump's automatic start setpoint is __(2)__ psig.

	__(1)__	__(2)__
A.	will	95
B.	will	110
C.	will NOT	95
D.✓	will NOT	110

K/A

086 **Fire Protection**

K4.02 **Knowledge of design feature(s) and/or interlock(s) which provide for the following:**

- Maintenance of fire header pressure.

K/A MATCH ANALYSIS

The question tests the candiate's knowledge of fire protection design features by asking if jockey pumps auto start on a trip of a pump and the auto start setpoint for the electric fire pump.

EXPLANATION OF REQUIRED KNOWLEDGE

The fire header pressure is normally maintained around 140 psig by operating 1 or 2 jockey pumps, depending on system leakage. Per ELEMENTARY CX3D-BH-F50C, the jockey pumps are manual start and stop only. If a leak in the fire system header or a sprinkler system actuation causes fire header pressure to drop, per SOP 13903-C

Limitation 2.2.9, the electric fire pump would start at 110 psig, Diesel fire pump #1 at 95 psig, and Diesel fire pump #2 at 85 psig. These three fire pumps have handswitches on the QPCP panel in the main control room that allow for remote starting and run status, but the pumps can only be stopped locally.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The jockey pumps are manual start and stop only. The remaining pumps in the system are designed to automatically start under the stated condition, and most of the pumps in the plant have automatic start features. It is reasonable to believe that the standby jockey pump would be designed to automatically start to maintain fire header pressure.

The second part is incorrect. The electric fire pump starts at 110 psig, Diesel fire pump #1 at 95 psig, and Diesel fire pump #2 at 85 psig. However, candidates routinely confuse the fire pump start setpoints.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. The electric fire pump would start at 110 psig.

C. Incorrect. Plausible. The first part is correct. The jockey pumps are manual start and stop only. Therefore, when the running jockey pump trips, the remaining jockey pump must be manually started.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T2 / G2	
K/A#	086K4.01	
Importance Rating:	3.0 / 3.4	
Technical Reference:	SOP 13903-C, Rev 44.0, page 6 ELEMENTARY CX3D-BH-F50C, Rev 3.0	
References provided:	None	
Learning Objective:	LO-PP-43101-05	Discuss the system response to an auto sprinkler actuation or fire hose operation which lowers system header pressure.
	LO-TA-43003	Abnormal and Emergency Starting Of A Diesel Fire Pump using 13903-C
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.7	
Comments:		

You have completed the test!

INITIALS

2.2.6 Per DOEJ-SM-C070400401-001, the Portable B.5.b Pump cannot be considered a backup fire suppression system as required by FP LCO 4.3 Cond A or B. The Portable B.5.b Pump is a defense in depth contingency for the interim period between total loss of suppression capability and arrival of Burke County EMA. A Burke County EMA pumper truck is the credited backup. ________

2.2.7 Per DOEJ-SM-C070400401-001, all hot work shall be suspended and hourly Fire Watches shall be established for 1A, 1B, 2A, and 2B Cable Spreading Rooms if a backup suppression system has to be established in either FP LCO 4.3 Cond A or B. ________

2.2.8 A Condition Report should be generated anytime two jockey pumps are required running to maintain Fire Protection System pressure. ________

2.2.9 Fire pumps auto start on decreasing header pressure at:

110 psig Electric fire pump starts ________

95 psig #1 Diesel fire pump starts ________

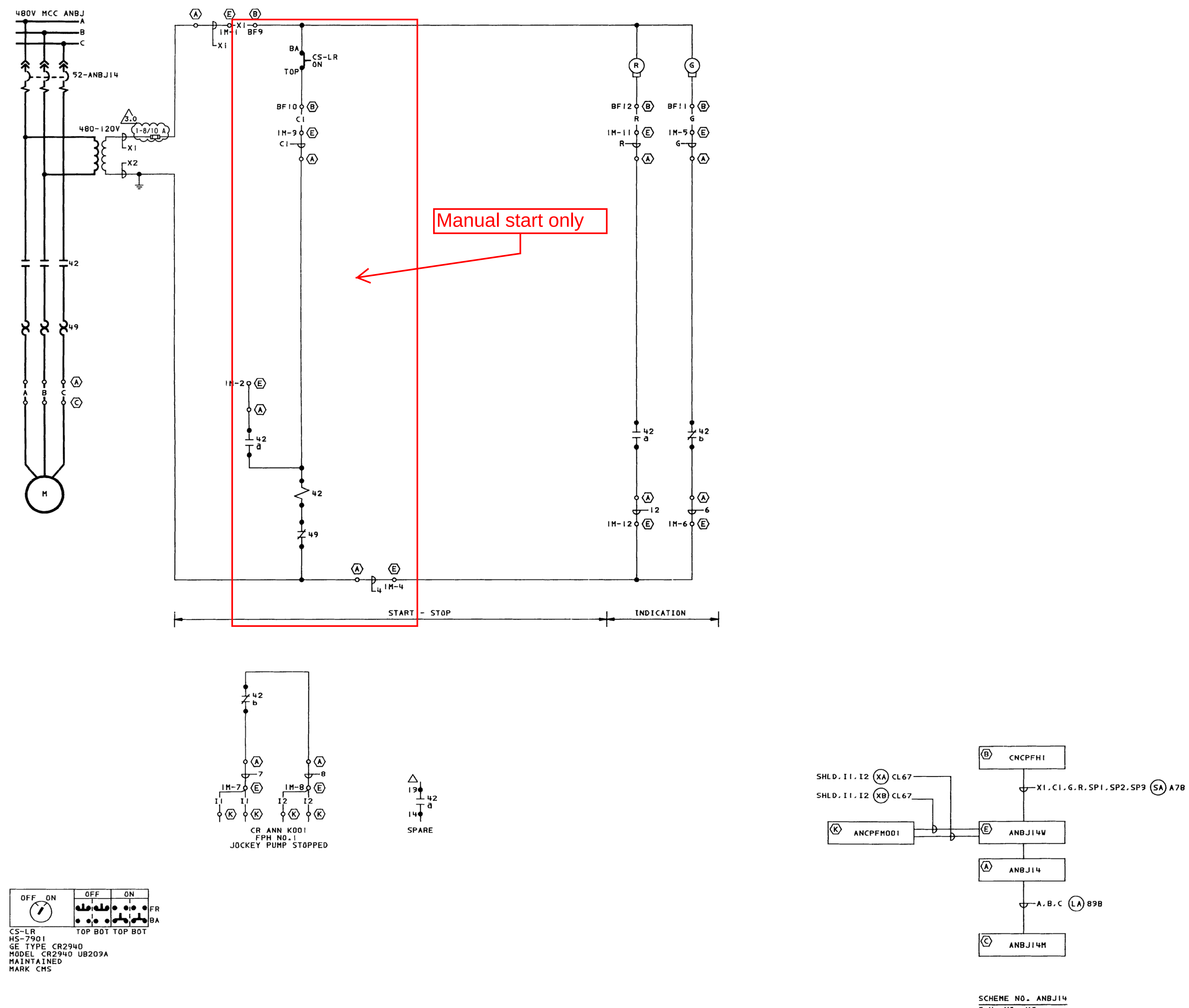
85 psig #2 Diesel fire pump starts ________

2.2.10 Attachment 1 provides guidance for developing troubleshooting steps in the event a low header pressure condition is encountered with no obvious explanation. A troubleshooting plan in accordance with NMP-AD-002 "Problem Solving and Troubleshooting Guidelines" must still be completed using these guidelines. (LVL 3 AI 2009201678) ________

3.0 PREREQUISITES

3.1 One or two Jockey Pump(s) is/are running to maintain or attempting to maintain the Fire Protection System pressure. ________

3.2 The Fire Pump House No. 1 and/or Fire Pump House No. 2 Heating, Ventilation, and Air Conditioning System (HVAC) has been aligned per 13330-C, "Outside Area Buildings HVAC System." ________



NOTES:

1. FOR SPECIAL NOTES & SYMBOLS REFER TO 0/L 1X3D-AA-A00A THRU A00Q.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER. FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
9. SPARE CONTACT WITH A "△" IS WIRED TO UNIT TERMINAL BLOCKS.

FPH NO.1 JOCKEY PUMP

NO.	VERSIONS	DATE	DR	CHK	APPV
3.0	REVISED PER ABN-V03026, VER. 1.0	6-22-12	JJB	SRG	AAN

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM - MISC SYS
FIRE PROTECTION WATER SYSTEM
C-2301-P4-001-M01

SCALE: NONE
JOB NO.10604
DRAWING NO. CX3D-BH-F50C
VER. 3.0
SIZE E 34X44
DWG CATEGORY: CRITICAL
CAD NAME: CXD-BH-F50C

1. 103A2.03 001/LOIT/RO/M/F 3.5/3.8/103A2.03/LO-TA-15003///

Initial conditions:

- LOCA has occurred on Unit 1.
- 19000-C, "Reactor Trip or Safety Injection," is entered.
- Verification of Immediate Operator Actions is complete.
- Containment pressure is 23.8 psig.

Current conditions:

- ALB06-D06 CNMT SPRAY ACTUATION is NOT LIT.
- ALB06-E06 CNMT ISO PHASE A ACTUATION is NOT LIT.

Which one of the following completes the following statement?

The **first** required action in 19000-C to be performed by the crew is to actuate __(1)__,

and

this action will be taken using __(2)__ direction.

	__(1)__	__(2)__
A.	CIA	Foldout Page
B.	CIA	OATC Initial Actions
C.✓	Containment Spray	Foldout Page
D.	Containment Spray	OATC Initial Actions

K/A

103 **Containment System**

A2.03 **Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.**

- Phase A and B isolation.

K/A MATCH ANALYSIS

The question tests the candidate's ability to predict the impact of high containment pressure on containment systems. The candidate is required to determine if CIA or Containment Spray actuation is the higher priority based on containment pressure, and, based on the priority, determine which procedure direction will accomplish the mitigating actions.

EXPLANATION OF REQUIRED KNOWLEDGE

Based on a containment pressure of 23.8 psig, both Containment Spray and CIA acutations signals should have been generated. Per 19000-C step 6 & 7, the Foldout Page actions are a higher priority than OATC/UO Initial Actions. EOP/AOP Writer's Guide 10020-C states that the Foldout Page is a continuous action page that is applicable to the entire procedure for which it is included. Therefore, any Foldout Page action is applicable and has higher priority than the OATC/UO Initial Actions.

Containment Spray Actuation is listed on Foldout Page step 3 and OATC Initial Actions step 8. CIA is not on the Foldout Page and is OATC Initial Action step 2. Therefore, with both actuations present, Containment Spray should be manually aligned <u>first</u> using the Foldout Page guidance.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Containment Spray would be manually aligned using the Foldout Page as the <u>first</u> action. However a candidate not familiar with the contents of the Foldout Page of 19000-C might believe CIA is also contained on the Foldout Page. As such, it would be reasonable to assume that CIA would be a higher priority than Containment Spray to ensure release to the public is minimized. This is the order of the items in the OATC Initial Actions.

The second part is correct. The Containment Spray actuation direction would be utilized from the Foldout Page.

B. Incorrect. Plausible. The both the first and second parts are incorrect. Containment Spray would be manually aligned using the Foldout Page as the <u>first</u> action. However a candidate who does not recognize the Containment Spray acutation setpoint has been exceeded and only believes the CIA actuation is present or who forgets about Containment Spray actuation being on the Foldout Page would believe that CIA would be the <u>first</u> system to align using step 2 of the OATC Initial Actions since CIA is not on the Foldout Page. Containment Spray would be verified at step 8.

C. Correct. The first part is correct. Containment Spray would be manually aligned using the Foldout Page as the <u>first</u> action.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. Containment Spray would be manually aligned using the Foldout Page. However, a candidate unfamiliar with 19000-C may recognize both Containment Spray and CIA actuations present but believe Containment Spray is listed first in the OATC Initail Actions and

believe alignment would be addressed using the body of the procedure instead of the Foldout Page or not remember Containment Spray being on the foldout page.

Level:	RO	
Tier # / Group #	T2/G1	
K/A#	103A2.03	
Importance Rating:	3.5/3.8	
Technical Reference:	EOP 19000-C, Rev 37.1, pages 5, 18, 20 & 35 ADMIN 10020-C, Rev 9.0, page 9 ARP 17006-1, Rev 33.1, pages 3, 42, 43, 52	
References provided:	None	
Learning Objective:	LO-LP-37002-01	State how each of the following EOP format elements are used to guide the operator in proper performance of the steps of the procedure. e. Foldout page
	LO-TA-15003	Manually Actuate and Align Containment Spray for Operation per 19000-C
	LO-TA-28016	Manually actuate CIA / CVI
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.5 / 41.7 / 43.5 / 45.3 / 45.13	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. Check if SI is actuated: • Any SI annunciator – LIT. • SI ACTUATED BPLB window - LIT. Go to Step 6.	4. Check if SI is required: <u>IF</u> one or more of the following conditions has occurred: • PRZR pressure less than or equal to 1870 psig. • Steam line pressure less than or equal to 585 psig. • Containment pressure greater than or equal to 3.8 psig. • Automatic alignment of ECCS equipment to injection phase. <u>THEN</u> actuate SI and go to Step 6.
<u>SUBSEQUENT OPERATOR ACTIONS</u>	
5. Perform the following to limit RCS cooldown:	
a. Check NR level in at least one SG greater than 10%.	a. Maintain AFW flow greater than 570 gpm and go to 19001-C, ES-0.1 REACTOR TRIP RESPONSE.
b. Reduce AFW flow.	
c. Go to 19001-C, ES-0.1 REACTOR TRIP RESPONSE.	
6. Initiate the Foldout Page.	
7. Perform the following: • OATC Initial Actions Page. • UO Initial Actions Page.	

OATC INITIAL ACTIONS

Sheet 2 of 4

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Check both trains of ECCS equipment - ALIGNING FOR INJECTION PHASE:	1. Actuate SI.
• MLB indication.	
2. Check Containment Isolation Phase A - ACTUATED:	2. Actuate CIA.
• CIA MLB indication.	IF valves do NOT close, THEN close valves.
3. Check ECCS Pumps and NCP status:	
a. CCPs - RUNNING.	a. Perform the following for available CCP(s): 1) Place alternate miniflow valve handswitch in ENABLE PTL: HS-8508A HS-8508B 2) Start CCP(s).
b. SI Pumps - RUNNING.	b. Start Pumps.
c. RHR Pumps - RUNNING.	c. Start Pumps.
d. NCP - TRIPPED.	d. Stop the NCP.
4. Verify CCW Pumps - ONLY TWO RUNNING EACH TRAIN.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. Check Containment pressure - REMAINED LESS THAN 21.5 PSIG.	8. Verify the following:
	a. Containment Spray actuated.
	b. Containment Spray Pump discharge valves open.
	c. Containment Spray Pumps running.
9. Check ECCS flows:	
a. BIT flow.	a. Align Valves using ATTACHMENT B.
b. RCS pressure - LESS THAN 1625 PSIG.	b. Go to Step 10.
c. SI Pump flow.	c. Align Valves using ATTACHMENT C.
d. RCS pressure - LESS THAN 300 PSIG.	d. Go to Step 10.
e. RHR Pump flow.	e. Align Valves using ATTACHMENT D.
10. Check ECCS Valve alignment - PROPER INJECTION LINEUP INDICATED ON MLBs.	10. Align valves using ATTACHMENT B, ATTACHMENT C, and ATTACHMENT D as necessary.

FOLDOUT

1. RCP TRIP CRITERIA
   Trip all RCPs if BOTH conditions listed below occur:

   a. CCPs or SI pumps - AT LEAST ONE RUNNING.

   b. RCP Trip Parameter - RCS PRESSURE LESS THAN 1375 PSIG.

2. AFW SUPPLY SWITCHOVER CRITERION
   Switch to alternate CST by initiating 13610, AUXILIARY FEEDWATER SYSTEM when CST level lowers to less than 15%.

3. CNMT SPRAY ACTUATION CRITERIA
   Verify the following when CNMT pressure is greater than or equal to 21.5 psig:

   a. CNMT Spray actuated.

   b. CNMT Spray Pump discharge valves open.

   c. CNMT Spray Pumps running.

4. Monitor SPENT FUEL POOL COOLING conditions:

   - Verify annunciators ALB05-A6, SPENT FUEL PIT HI TEMP and ALB05-E2, SPENT FUEL PIT LOW LEVEL are both clear.
     IF alarms are NOT CLEAR,
     THEN initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING.

   - IF SPENT FUEL POOL LEVEL OR COOLING ALARMS are not available,
     THEN dispatch operator to start 2 HR interval local checking that level > 217 ft and temperature < 130°F.
     IF either parameter is exceeded,
     THEN initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING

   - IF applicable, Using PTDB TAB 26, determine time to restore SFP LEVEL OR COOLING is < time to reach 200°F in Spent Fuel Pool.
     IF NOT initiate 18030-C, LOSS OF SPENT FUEL POOL LEVEL OR COOLING

	(1)	(2)	(3)	(4)	(5)	(6)
A	RHR PMP 1 DISCH HI PRESS		ACCUM TANK 1 HI/LO LEVEL	ACCUM TANK 1 HI/LO PRESS	ACCUM TANK 1 ISO VLV 8808A NOT FULLY OPEN	CNMT HI-1 PRESS ALERT ADVERSE CNMT
B	RHR PMP-2 DISCH HI PRESS	RCS MIDLOOP HI LEVEL	ACCUM TANK 2 HI/LO LEVEL	ACCUM TANK 2 HI/LO PRESS	ACCUM TANK 2 ISO VLV 8808B NOT FULLY OPEN	CNMT HI-2 PRESS ALERT
C	RHR PMP OVERLOAD TRIP	RCS MIDLOOP LO LEVEL	ACCUM TANK 3 HI/LO LEVEL	ACCUM TANK 3 HI/LO PRESS	ACCUM TANK 3 ISO VLV 8808C NOT FULLY OPEN	CNMT HI-3 PRESS ALERT
D	RHR HL VLV OPEN AND HI RCS PRESS		ACCUM TANK 4 HI/LO LEVEL	ACCUM TANK 4 HI/LO PRESS	ACCUM TANK 4 ISO VLV 8808D NOT FULLY OPEN	CNMT SPRAY ACTUATION
E	CNMT VENT ISO ACTUATION		RWST TO SI PMP ISO VLV 8806 NOT FULLY OPEN	RWST LO LEVEL	RWST EMPTY LEVEL	CNMT ISO PHASE A ACTUATION
F	CSFST TROUBLE		RWST HI LEVEL	RWST LO-LO LEVEL	RWST LO-LO LEVEL ALERT	SI PMP OVERLOAD TRIP

WINDOW D06

CNMT SPRAY
ACTUATION

ORIGIN	SETPOINT
2 out of 4 1-PT-0934A 1-PT-0935A 1-PT-0936A 1-PT-0937A	21.5 psig (2/4 channels) (relay K643)
or both 1-HS-40010 1-HS-40011	Not Applicable
or both 1-HS-40004 1-HS-40005	Not Applicable

1.0 PROBABLE CAUSE

1. Manual actuation of the Containment Spray System.
2. Containment HI-3 setpoint reached on 2 or more Containment pressure channels.

2.0 AUTOMATIC ACTIONS

1. Containment Spray Pumps start.
2. Containment Spray Isolation Valves 1-HV-9001A and 1-HV-9001B open.

3.0 INITIAL OPERATOR ACTIONS

NOTE

Actions for a containment spray actuation are contained in Emergency Operating Procedures.

WINDOW D06
(Continued)

4.0 SUBSEQUENT OPERATOR ACTIONS

IF a spurious Containment Spray actuation has occurred:

1. **Reset** CS signal.

2. **Stop** CS pumps.

3. **Shut** CNMT SPRAY ISO VLVS:

   a. 1-HV-9001A

   b. 1-HV-9001B

4. IF Containment Spray is actuated and terminated prior to recirculation, a controlled cleanup and inspection of equipment in containment should begin within five days of the event

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB131, 1X6AA02-232, PLS, 1X6AU01-178, 1X6AX01-322, 1X6AX01-409, 1X6AX01428, 1X6AV01-242

WINDOW E06

CNMT ISO
PHASE A
ACTUATION

ORIGIN	**SETPOINT**
Safety Injection	Not Applicable
OR 1-HS-40006 or 1-HS-40009	Not Applicable

>3.8 psig

1.0 PROBABLE CAUSE

1. Safety Injection Actuation
2. Manual Actuation.

2.0 AUTOMATIC ACTIONS

Initiates Containment Phase A Isolation.

3.0 INITIAL OPERATOR ACTIONS

If a Safety Injection has occurred, **initiate** 19000-C, "E-O Reactor Trip Or Safety Injection."

4.0 SUBSEQUENT OPERATOR ACTIONS

1. IF an inadvertent Phase A Isolation has occurred in Modes 1, 2 or 3, then **perform** the following:

   a. **Reset** Phase A by placing both 1-HS-40120 and 1-HS-40122 to RESET position,

   b. **Open** Instrument Air to containment 1-HV-9378 using both 1-HS-9378A and 1-HS-9378B,

   c. **Restore** normal letdown/charging per 13006-1, "Chemical And Volume Control System,"

   d. **Open** RCP Seal Return 1-HV-8100 and 1-HV-8112 using 1-HS-8100 and 1-HS-8112,

   e. **Reset** Containment Ventilation Isolation by placing both 1-HS-40121 and 1-HS-40123 to RESET position.

2. **Complete** the applicable portions of 11886-1, "Recovery From ESF Actuations," for CIA and CVI.

3.5 CONTINUOUS ACTION STEPS

3.5.1 Continuous Action steps are marked in EOPs and AOPs by bolding the AER step and by use of an asterisk (*) immediately preceding the AER step number (or RNO step number if there is one). Only the high level step number is marked with the asterisk (*) indicating that there is a continuous action in the AER or RNO text, the associated Note, or Caution.

3.5.1.1 Continuous action steps that are "skipped" by procedure step transitions, are still in affect and should be applied.

3.5.1.2 Continuous Action Pages will be posted in a separate book in the control room for EOPS and AOPs that have continuous actions and updated as new revisions are issued.

3.5.2 NOTES and CAUTIONS may contain continuous action steps if written passively.

3.5.3 Logical terms such as WHEN, IF, THEN are considered continuous operator actions.

3.5.4 Action verbs such as control, limit, match, maintain, and monitor are continuous action verbs. See TABLE 2 for definitions of these verbs.

3.5.5 NOTEs and CAUTIONs prior to step 1 of EOPs and AOPs apply to the entire procedure.

3.5.6 Foldout Page is a continuous action page that is applicable to the entire procedure for which it is included.

3.5.6.1 Foldout Pages will be posted in a separate book (the same book as the continuous actions) in the control room for EOPS that have foldout pages.

3.5.6.2 The foldout page should be copied on the back of each page in the EOP except for the foldout page itself.

3.5.7 Continuous action steps are no longer applicable when exiting a procedure unless restated in the next procedure entered, or they direct return to the procedure in which they are stated, e.g. 19013-C where you are directed to return and perform Containment Spray Pump suction swap once conditions are met.

1. 103K1.08 001/LOIT/RO/M/F 3.6/3.8/103K1.08/LO-PP-28103-06///

Given the following:

- A LOCA with SI actuation occurred on Unit 1.
- SI can NOT be reset.

Which one of the following completes the following statement?

The CIA signal __(1)__ be reset,

and

the CVI signal __(2)__ be reset.

	__(1)__	__(2)__
A.✓	can	can
B.	can	can NOT
C.	can NOT	can
D.	can NOT	can NOT

K/A

103 **Containment System**

K1.08 **Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems:**

- SIS, including action of safety injection reset

K/A MATCH ANALYSIS

Question requires the candidate to determine which interlocks are required to be cleared or reset in order to reset CIA and CVI, and is based on RO knowledge level requirements. In both cases the candidate must determine if the logic device is "retentive memory" or "retentive memory with actuation block" when making a determination of system response.

EXPLANATION OF REQUIRED KNOWLEDGE

In the EOP network, the reset of either CIA or CVI signal is always preceeded with a step that resets SI. (See 19012-C step 2 and 3, for example.) The RNO for resetting SI directs the operator to perform an Attachment that overrides SSPS and manually rolls the slave relays and effectively resets SI. Per LOGIC 1X6AA02-00232, both CVI

and CIA can be reset with the SI actuation signal present. Once blocked, the block will hold and prevent any subsequent actuation signal from being processed. Therefore, the inability to reset SI will not prevent the reset of either CIA or CVI and does not impede the progress of the procedure. The inability to reset SI may complicate other recovery actions later and therefore it is desirable to reset SI using the Attachment. (Note: This RNO is a relatively new addtion to the EOPs based on an industry event in which an inadvertant SI was encountered and could not be reset, resulting in challenges to the operating crew. Even without SI reset, the plant is capable of being maintained. However, knowledge space decisions are required. This RNO puts operation in rule space and allows normal and expected responses to EOP steps.)

ANSWER / DISTRACTOR ANALYSIS

A. Correct.

The first part is correct. The CIA reset logic device is a "retentive memory with actuation block" which is designed to allow the actuation signal to be overriden. This knowledge is important because procedures are designed in general to reset the initiating signal (such as 'SI') first. However, if the ability to reset 'SI' has failed, the operator must know that this does not impede the ability to perform reset of CIA.

The second part is correct. The CVI reset logic device is a "retentive memory with actuation block" which is designed to allow the actuation signal to be overriden. This knowledge is important because procedures are designed in general to reset the initiating signal (such as 'SI') first. However, if the ability the reset 'SI' has failed, the operator must know that this does not impede the ability to reset of CVI.

B. Incorrect. Plausible.

The first part is correct. See the first part of choice A above.

The second part is incorrect. The CVI reset logic device is a "retentive memory with actuation block" which is designed to allow the actuation signal to be overriden. However, there is no direction within the E-1/ES-1 series of procedures that directs the resetting of CVI. Therefore, it is reasonable for a candidate to conclude that "retentive memory with actuation block" would not be necessary for this actuation signal and SI must be reset prior to CVI being reset.

C. IncorrectPlausible.

The first part is incorrect. The CIA reset logic device is a, "retentive memory with actuation block" which is designed to allow the actuation signal to be overriden. However the candidate may determine the CIA actuation signal uses the standard "retentive memory" which has no actuation block capability and SI must be reset first. This logic would be erroneously re-enforced by the procedural sequence and RNO as described above in the Explanation of Required Knowledge.

The second part is correct. See the second part of choice A

above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level: RO
Tier # / Group # T2 / G1
K/A# 103K1.08
Importance Rating: 3.6 / 3.8

Technical Reference: LOGIC 1X6AA02-00232, Rev
EOP 19012-C, Rev 33.3, pages 3 & 39

References provided: None

Learning Objective: LO-PP-28103-06 Determine when ESF actuation signal can be reset and describe actions required to reset the signal.
LO-PP-28103-07 Discuss SI reset to include:
a. Time delay
b. SI reset with P-4
c. SI reset without P-4
d. Auto and Manual actuation capabilities following reset

Question origin: NEW

Cognitive Level: M/F

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments:

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Initiate the following: • Continuous Actions and Foldout Page. • Critical Safety Function Status Trees per 19200-C, F-O CRITICAL SAFETY FUNCTION STATUS TREE.	

CAUTION

If offsite power is lost after SI reset, action is required to restart the following ESF equipment if plant conditions require their operation:

- RHR Pumps
- SI Pumps
- Post-LOCA Cavity Purge Units
- Containment Coolers in low speed (Started in high speed on a UV signal).
- ESF Chilled Water Pumps (If CRI is reset).

2. Reset SI.	2. IF SI will NOT reset, THEN initiate ATTACHMENT D.

CAUTION

Repositioning Phase A Isolation Valves may cause radiation problems throughout the plant.

3. Reset Containment Isolation Phase A.

ATTACHMENT D

Sheet 1 of 1

RESPONSE TO INADVERTENT SI AND INABILITY TO RESET OR BLOCK SI

1. Identify the affected train. Circle: A Train B Train

NOTE

De-energizing the two 48 VDC power supplies to a train of SSPS will result in the following:

- General Warning
- ALB05-E06 or ALB05-F06 will illuminate
- Undervoltage Driver output de-energizes
- Reactor Trip condition (Reactor Trip Breaker OPEN) on the affected train (already initiated from the Turbine Trip)
- 48 VDC is removed from all master relays

2. At the affected train SSPS Logic Cabinet, de-energize both 48 VDC power supplies (Located in the upper 2 sections) by placing the ON/OFF switch to the OFF position.

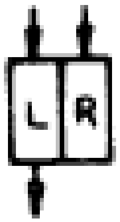
3. At the affected train Safeguards Test Cabinet (STC) #1, reset SSPS Slave Relays by momentarily turning TEST RESET SWITCH S-821 to the RESET position.

4. At the affected train Safeguards Test Cabinet (STC) #2, reset SSPS Slave Relays by momentarily turning TEST RESET SWITCH S-921 to the RESET position.

5. At the affected train, locate and open the Output Cabinet and place the MODE SELECTOR Switch in the TEST position and check the OPERATE lamp NOT lit.

6. Notify I&C to investigate the affected train SSPS to determine the source of the SI signal.

° END OF ATTACHMENT D

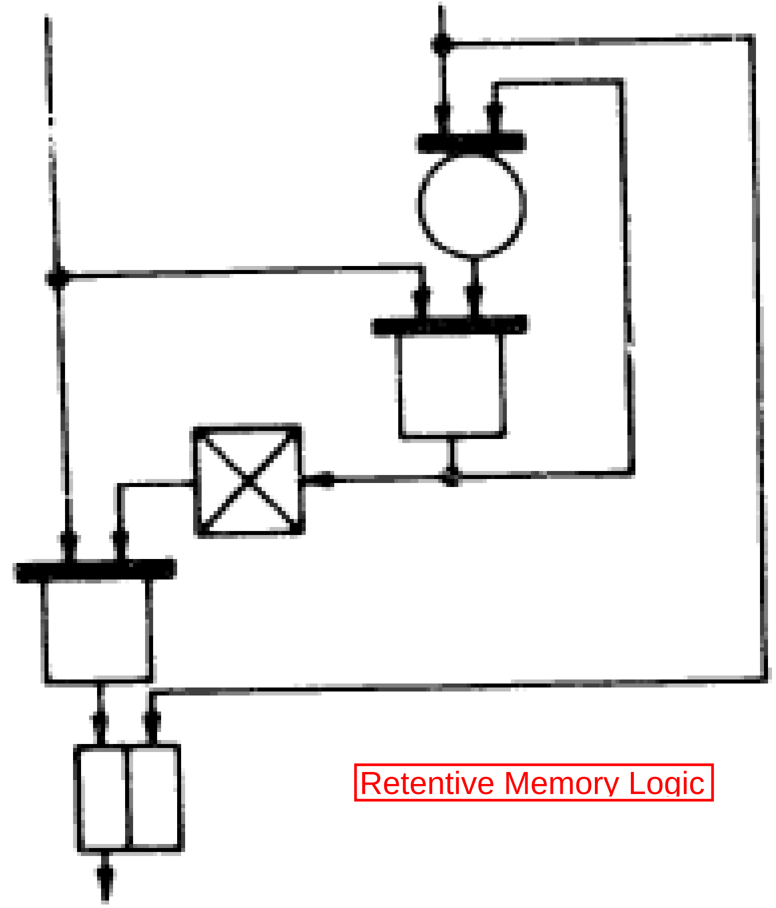


RETENTIVE MEMORY WITH ACTUATION BLOCK

A DEVICE HAVING RETENTIVE MEMORY AND ACTUATION SIGNAL BLOCK LOGIC FUNCTIONS AS INDICATED BY THE DIAGRAM BELOW.

ACTUATING SIGNAL

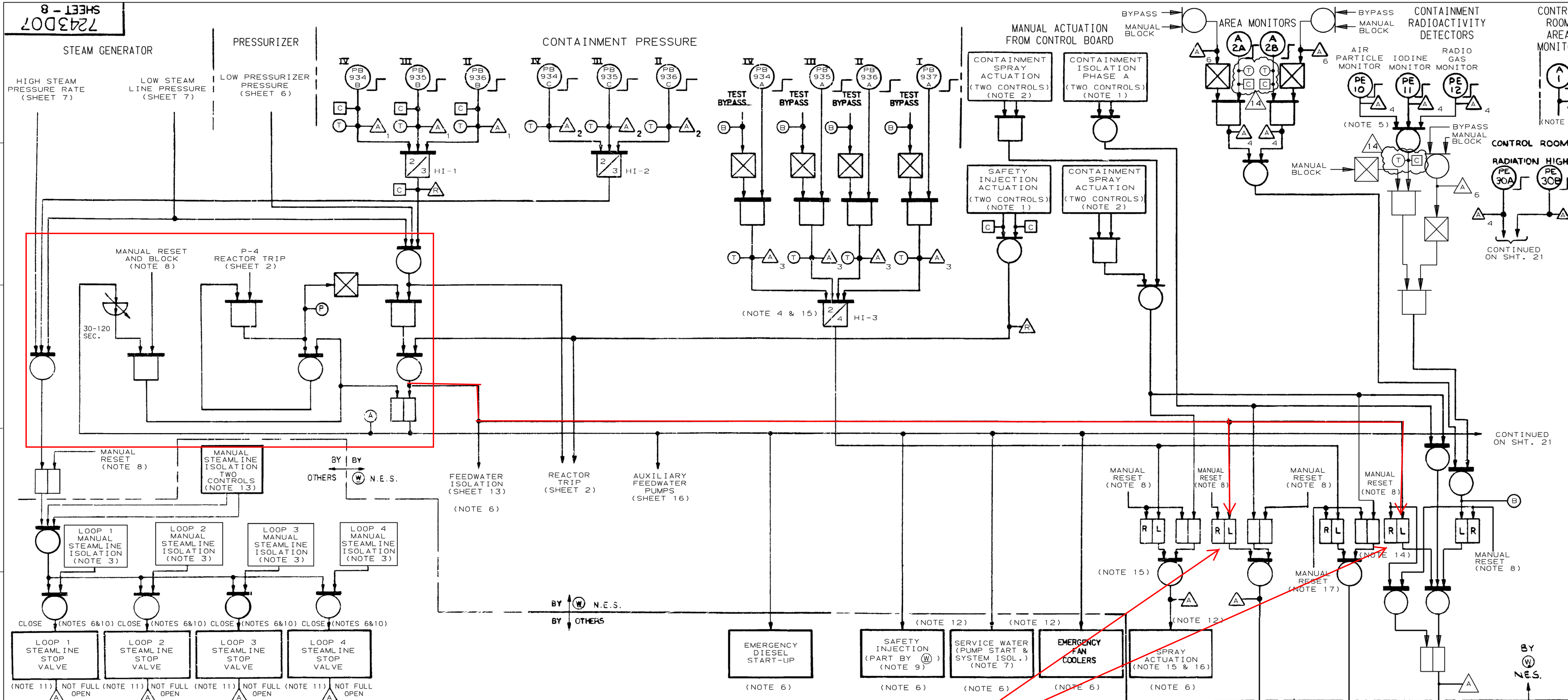
RESET (MOMENTARY)

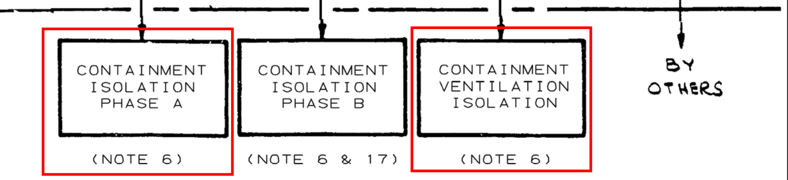


OUTPUT SIGNAL

Retentive Memory Logic

ANALOG





NOTES:

1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, OPERATING EITHER CONTROL WILL ACTUATE.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS, ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY.
3. ONE CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE-TO-ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. CIRCUITRY IS NOT PART OF THE SAFEGUARDS SYSTEM, AND IS NOT SAFETY RELATED, HOWEVER, CIRCUITRY IS INPUT TO THE SAFEGUARS SYSTEM THERFORE ISOLATION MUST BE PROVIDED.
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. SERVICE WATER SYSTEM ISOLATION IS USED ONLY IF REQUIRED.
8. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
9. SAFETY INJECTION SEQUENCE REQUIREMENTS (IF SEQUENCING IS NECESSARY) ARE SPECIFIED BY Ⓦ NUCLEAR ENERGY SYSTEMS.
10. ALSO CLOSES THE BYPASS VALVE IN PARALLEL WITH THE ASSOCIATED STEAM LINE STOP VALVE.
11. LIGHTS SHOULD BE PROVIDED IN THE CONTROL ROOM FOR EACH STEAM LINE STOP VALVE TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED, MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF Ⓦ NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE Ⓦ SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY, AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE SEQUENCE TO LOAD THE CONTAINMENT SPRAY SYSTEM ON THE EMERGENCY DIESEL POWER SHOULD BE INDEPENDENT OF THE SEQUENCE TO LOAD THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY THE SEQUENCE INTERLOCK SHOULD BE SUCH THAT SPRAY WILL START WITHIN THE REQUIRED TIME INDEPENDENT OF THE SAFETY INJECTION SIGNAL STATUS.
17. CONTAINMENT ISOLATION PHASE B IS NOT USED. THE INITIATION LOGIC AND MANUAL RESET ACTION FOR CONTAINMENT ISOLATION PHASE B SHOWN ON THIS SHEET IS FOR INFORMATION ONLY.

NO.	REVISIONS	DATE	DR	CHK	APPV
14	INCORPORATED PER DCP 97-V1N0067	11-16-00	ELC	TSL	GLB

SCS REVISIONS

DOCUMENT STATUS CODE 1

1X6AA02-00232-17

NEXT ASSY. REF. DWG.

DFTM. R HEARN 5/10/73
CHKR.
DES. ENG. JV Gianaio 7/12/73
MFG. ENG.
MTLS. ENG.
APP. D G Patton 7-12-73
APP.
APP.
DFTG. SUPV. 7-13-73

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA., U.S.A.

TITLE: GEORGIA POWER CO.
ALVIN W. VOGTLE UNITS 1 & 2
FUNCTIONAL DIAGRAM
SAFEGUARD ACTUATION SYSTEM

SCALE
DIMENSIONS IN INCHES
DO NOT SCALE

7243D07
SHEET - 8
SUB 13

1. G2.1.14 001/LOIT/RO/M/F 2.7/4.1/G2.1.14/LO-LP-05110-01//HL-15R/

Which one of the following completes the following statement?

Plant announcements are required per UOP guidance for reactor __(1)__, and per EOP guidance when realigning ECCS for __(2)__ leg recirculation.

	__(1)__	__(2)__
A.	startups	hot
B.✓	startups	cold
C.	shutdowns	hot
D.	shutdowns	cold

K/A

G2.1.14 Generic

Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes,etc.

K/A MATCH ANALYSIS

The question requires the student to correctly identify which plant evolutions require a Plant PA anouncement (notifying internal organizations).

EXPLANATION OF REQUIRED KNOWLEDGE

Per UOP 12003-C step 4.2.12, announcement of the reactor startup to plant personnel is required prior to commencement of rod withdrawal or dilution to criticality in steps 4.2.13 and 4.2.14. In contrast, UOP 12005-C does not contain any direction to make page announcements during a reactor shutdown.

Per EOP 19013-C step 5, operators are directed to make a page announcement to clear personnel from the Auxiliary Building prior to initiating Cold Leg Recirc. In contrast, EOP 19014-C does not contain any diretion to make page announcements prior to transition to Hot Leg Recirc.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per UOP 12003-C step 4.2.12, announcement of the reactor startup to plant personnel is required prior to commencement of rod withdrawal or dilution to criticality in steps 4.2.13 and 4.2.14.

The second part is incorrect. EOP 19014-C does not contain

any direction to make page announcements prior to transition to Hot Leg Recirc. However, EOP 19013-C step 5 directs operators to make a page announcement to clear personnel from the Auxiliary Building prior to initiating Cold Leg Recirc. It is reasonable for a candidate to confuse the two procedures and believe 19013-C is for Hot Leg Recirc.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. EOP 19013-C step 5 directs operators to make a page announcement to clear personnel form the Auxiliary Building prior to initiating Cold Leg Recirc.

C. Incorrect. Plausible. The first part is incorrect. UOP 12005-C does not contain any direction to make page announcements during a reactor shutdown. However, candidates are trained to make a page announcement of an unplanned Reactor Trip and/or Safety Injection. This is not a procedural requirement, but a good practice. It is reasonable for a candidate to assume that since an unplanned Reactor Trip is announced, any reactor trip is required to be announced.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.1.14	
Importance Rating:	3.1 / 3.1	
Technical Reference:	UOP 12003-C, Rev 51.2, pages 25 thru 27 UOP 12005-C, Rev 28.0 EOP 19013-C, Rev 29.2, page 5 EOP 19014-C, Rev 15.2	
References provided:	None	
Learning Objective:	LO-LP-05110-01	Describe the functions of the following and explain how and when each of the following are used: a. Telephone/Page System
	LO-TA-13009	Manually align ECCS for Cold Leg Recirculation Phase using EOP 19013-C
	LO-TA-13012	Manually align ECCS for Hot Leg Recirculation Phase using EOP 19014-C
	LO-TA-61001	Reactor Startup using 12003-C
	LO-TA-61003	Reactor Shutdown to Hot Standby using 12005-C
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.10 / 43.5 / 45.12	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. Initiate ATTACHMENT A to align ECCS Pumps to the Cold Leg Recirculation flowpath and continue with Step 4.	
4. Notify Health Physics that radiation levels in the Auxiliary Building will change when Cold Leg Recirculation is established.	
5. Make a page announcement to clear personnel from the Auxiliary Building prior to initiating Cold Leg Recirculation.	
6. Initiate Continuous Actions Page.	
*7. **Check RWST level – GREATER THAN 8%.**	*7. Stop any ECCS Pumps taking suction from the RWST.
*8. **Check if SI pumps should be stopped.**	
a. RCS pressure - GREATER THAN 1625 PSIG.	a. IF RCS pressure rises to greater than 1625 psig, THEN stop SI Pumps. Go To Step 9.
b. Stop SI Pumps.	
9. Check ATTACHMENT A - COMPLETE.	9. Do NOT continue with this procedure until ATTACHMENT A has been COMPLETED.

INITIALS

NOTE

Step 4.2.11 is not applicable if the conditions of Step 4.2.10 have been applied. ☐

4.2.11 **Initiate** an Inverse Count Rate Ratio (ICRR) plot per Data Sheet 1. (1987213952, 1988214388, 1988214835, 1989215543) ________

CAUTIONS

- All pertinent indications should be monitored during approach to criticality such as flux level, SUR, recorders, and count rate instrumentation.(19843001331, 1990318409) ☐
- Conservative action should be taken, (i.e., stop startup and insert control rods), whenever unexpected situations arise, with respect to reactivity, criticality, power level, or any other anomalous behavior of the reactor core. ☐
- Activities should be avoided during reactor startup that could distract operators and supervisors involved with the startup such as shift turnovers and surveillance testing. (1990318409) ☐

4.2.12 **Announce** the reactor startup to plant personnel. ________

INITIALS

NOTES

- IF the startup has been delayed since completing Step 4.2.9, **repeat** Step 4.2.9 prior to commencing the startup per Step 4.2.13. ☐
- IF Reactor startup is suspended for shift turnover, Step 4.2.13 should be reperformed, prior to proceeding. ☐
- IF this startup is a dilution to criticality for LPPT, Step 4.2.13 should be performed in conjunction with LPPT-GAE/GBE-01. ☐

CAUTIONS

- Criticality SHALL be anticipated any time positive reactivity is being added, including but NOT limited to when control rods are being withdrawn or RCS dilution in progress. ☐
- A sustained SUR of 1.0 dpm should not be exceeded. ☐
- During approach to criticality, two positive reactivity additions will NOT be performed simultaneously. ☐

4.2.13 **Commence** Rod Withdrawal:

a. **Verify** shutdown banks withdrawn greater than or equal to the insertion limit specified in the COLR. (TS 3.1.5) (1987212898) ________

b. **Verify** Rod Bank Selector Switch in MANUAL. ________

____
IV

c. **Withdraw** rods in 50 step increments, or as directed by the Reactor Engineer and approved by the SS. ________

INITIALS

NOTE

For startups following a refueling outage where initial criticality will be achieved by dilution to critical for Low Power Physics testing, Mode 2 shall be declared when dilution commences. ☐

4.2.14 WHEN Control Bank withdrawal commences, OR dilution to Criticality for LPPT commences at Step 4.2.21.b, whichever is applicable, **perform** the following (1984300231, 19853003137):

a. **Log** Mode 2 entry into the Control Room Log

Record Time: ________ ________

b. **Update** IPC to reflect Mode as follows:

(1) On Top Right of IPC Screen **click** on MODE. ________

(2) **Click** on the desired set mode button. ________

(3) **Verify** current mode changes to the desired mode. ________

4.2.15 WHEN Control Bank A reaches 12 steps as indicated by DRPI, **verify** the following annunciators reset:

a. ROD AT BOTTOM (ALB10E05), ________

b. TWO OR MORE RODS AT BOTTOM (ALB10F05). ________

4.2.16 WHEN Control Bank A reaches 115 steps, **verify** Control Bank B begins withdrawing. (TS SR 3.1.6.3) (1995330520) ________

4.2.17 WHEN Control Bank B reaches 115 steps, **verify** Control Bank C begins withdrawing. (TS SR 3.1.6.3) (1995330520) ________

1. G2.1.28 001/LOIT/RO/M/F 4.1/4.1/G2.1.28/LO-PP-23101-09///

Initial condition:

- Unit 1 is at 100% reactor power.

Current conditions:

- ARE-2532A, FHB - Effluent Radiogas, fails high.
- ALB05-C03 HIGH RADIATION ALARM is received.

Which one of the following completes the following statement?

When ALB05-C03 was received, __(1)__ FHB Post Accident Filter Unit(s) automatically started,

and

when the FHB Isolation Reset handswitch is taken to RESET, the white light illuminating alerts the operator that the actuation signal __(2)__ present.

	__(1)__	__(2)__
A.	ONLY one	is
B.	ONLY one	is NOT
C.✓	BOTH	is
D.	BOTH	is NOT

K/A

G2.1.28 Knowledge of the purpose and function of major system components and controls.

K/A MATCH ANALYSIS

The KA addresses the purpose of major components and controls. The question has identified the Fuel Handling Building (FHB) HVAC system as the major component and the white handswitch lights as a system function designed to allow the operator to analyze system response. In addition, the candidate is given a specific event and must determine system response and control indications in order to make a correct decision.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17102-1, ARE-2532A in HIGH alarm would switch the Normal Fuel Handling Building Ventilation to Accident Mode Ventilation. Per SOP 13320-1 step 4.2.2, when a FHB actuation is received from either train, BOTH FHB Post-Accident Filter Fans start.

Per SOP 13320-1 Precaution 2.1.4, if a FHB Isolation signal is still present and either train's hand switch is taken to RESET, then that train's isolation logic is rendered inoperable and the corresponding WHITE light will be LIT on AHS-2532B/2533B.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per SOP 13320-1 step 4.2.2, when a FHB actuation is received for either train, BOTH FHB Post-Accidnet Filter Fans start. However, the candidate may determine that since only the Train A powered radiation monitor went into alarm only the Train A unit will automatically start.

The second part is correct. Per SOP 13320-1 Precaution 2.1.4, if a FHB Isolation signal is still present and either train's hand switch is takent to RESET, then that train's isolation logic is rendered inoperable and the corresponding WHITE light will be LIT on AHS-2532B/2533B.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per SOP 13320-1 Precaution 2.1.4, if a FHB Isolation signal is still present and either train's hand switch is takent to RESET, then that train's isolation logic is rendered inoperable and the corresponding WHITE light will be LIT on AHS-2532B/2533B. However, the candidate may determine that the white light indicates the signal has been reset and manual alignment is allowed because the signal is no longer present.

C. Correct. The first part is correct. Per SOP 13320-1 step 4.2.2, when a FHB actuation is received for either train, BOTH FHB Post-Accident Filter Fans start.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.1.28	
Importance Rating:	4.1 / 4.1	
Technical Reference:	SOP 13320-C, Rev 33.1, pages 4 & 16 ARP 17102-1, Rev 20.2, page 25	
References provided:	None	
Learning Objective:	LO-PP-23101-09	Explain how the Fuel Handling Building HVAC System responds to a Fuel Handling Building Isolation Signal.
	LO-PP-23101-07	Describe the Fuel Handling Building HVAC System flow path for both Normal and Post Accident Conditions.
Question origin:	MODIFIED - HL18 NRC # 072K1.03	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.7 / 41.13	
Comments:		

You have completed the test!

1. 072K1.03 001/2/2/ARM - FHBI/C/A - 3.6/3.7/MOD - HL-15R AUDIT/HL-18 NRC/RO/SRO/KAJ

A dropped spent fuel assembly in the Unit 1 Spent Fuel Pool has resulted in the following radiation monitor alarms:

- 1-RE-0008, FHB Area Monitor, indicates HIGH.

- A-RE-2532A(B) and A-RE-2533A(B), FHB Effluent Monitors, indicate ALERT.

- The crew is implementing 18006-C, "Fuel Handling Event".

For the given conditions, which ONE of the following completes the following statement?

1-RE-0008 ___(1)___ provide audible and visual indications of the alarm in the Unit 1 SFP area,

and

the FHB Post-Accident Filtration Units ___(2)___ automatically start.

A. (1) will

(2) will

B.✓ (1) will

(2) will NOT

C. (1) will NOT

(2) will

D. (1) will NOT

(2) will NOT

Original Question

INITIALS

2.0 PRECAUTIONS AND LIMITATIONS

2.1 PRECAUTIONS

2.1.1 The Emergency Filtration Units automatically start on high radiation in the Exhaust Header. ________

2.1.2 Outside Air Intake Low Temperature Cutout Control Switch, A-TSL-2520 for A-1541-A7-001 or A-TSL-2521 for A-1541-A7-002 will trip the FHB Normal Supply Fan if it senses a temperature of less than or equal to 37°F. The switch must be manually reset, but will not reset until sensing bulb temperature increases to 49°F. The reset switches are located on the side of the corresponding HVAC supply unit. ________

2.1.3 One train of FHB Post Accident Filter Unit should be placed in service if the normal supply units will not operate due to low outside temperature as discussed in Precaution Step 2.1.2. ________

2.1.4 If a FHB Isolation Signal is still present and either train's hand switch is taken to reset, then that Train's Isolation logic is rendered inoperable. A corresponding White Light will be LIT on AHS-2532B(AHS-2533B), and a corresponding alarm is received on the Unit One and Unit Two SSMP Panels and bring in annunciators on Both Unit One and Unit Two (1/2ALB04-E01(E02). ________

2.1.5 Sudden changes in the HVAC configuration such as a CVI or FHB Isolation can cause rapid changes in Spent Fuel Pool (SFP) and Refueling Cavity water level when the two are interconnected. SFP and Refueling Cavity levels should be checked following any such change to prevent overflow. (1992224401) ________

2.2 LIMITATIONS

2.2.1 Two independent FHB Post Accident Exhaust Systems shall be operable whenever irradiated fuel is in either Spent Fuel Pool. (TR 13.9.5) ________

2.2.2 The Fuel Handling building Post Accident Ventilation Actuation Instrumentation shall be OPERABLE. (TR 13.3.6, Table 13.3.6-1) ________

INITIALS

CAUTION

The Train B Post Accident Filter Unit and the normal exhaust HVAC system discharge to the same exhaust header. They should not be aligned to discharge to their common exhaust stack at the same time. ☐

4.2.1 **Actuate** FHB ISOLATION by momentarily placing either train's handswitch to ACTUATE position.

FHB ISOLATION MANUAL ACTUATION	AHS-2532A (A54)	________
FHB ISOLATION MANUAL ACTUATION	AHS-2533A (A55)	________

4.2.2 **Verify** FHB Isolation:

a. FHB Isolation actuated:

- Red Light at FHB ISOLATION MANUAL ACTUATION, AHS-2532A (A54) LIT ________
- Red Light at FHB ISOLATION MANUAL ACTUATION, AHS-2533A (A55) LIT ________

b. POST ACCIDENT FILT/EXH FANS:

Train A: A-1542-N7-001 (C54)	RUNNING	________
Train B: A-1542-N7-002 (C55)	RUNNING	________

WINDOW CDCA B6

ORIGIN	SETPOINT	
Fuel Handling Building Effluent Radiogas Monitor ARE-2532A	As determined by Chemistry Department	A-RE-2532A (RED LAMP LIT) (HIGH)

NOTE

For other than HIGH conditions see Pages 5 and 6. ☐

1.0 **PROBABLE CAUSE**

1. High airborne radioactivity in the Fuel Handling Building.
2. Equipment malfunction.

2.0 **AUTOMATIC ACTIONS**

Switches the Normal Fuel Handling Building Ventilation to Accident Mode Ventilation.

3.0 **INITIAL OPERATOR ACTIONS**

Evacuate the Fuel Handling Building.

1. G2.1.8 001/LOIT/RO/M/F 3.4/4.1/G2.1.8/LO-TA-60047//H-17 NRC/

Given the following:

- Unit 1 Control Room is being evacuated due to a fire.
- The operating crew is fully staffed.

Which one of the following completes the following statement?

In accordance with 18038-1, "Operation From Remote Shutdown Panels," the __(1)__ will be dispatched to Shutdown Panel 'C' (TDAFW Pump Room),

and

the PREFERRED method of communications to coordinate in-plant activities with personnel outside the control room is via __(2)__.

	__(1)__	__(2)__
A.	Unit Operator	sound powered telephones (red box)
B.	Unit Operator	bridge phone extension 3145, codes 123# or 234#
C.✓	System Operator	sound powered telephones (red box)
D.	System Operator	bridge phone extension 3145, codes 123# or 234#

K/A

G.2.1.8 Ability to coordinate personnel activities outside the control room.

K/A MATCH ANALYSIS

The question presents a plausible scenario in which a Control Room evacuation is in progress. The candidate must determine which operator is required to report to Shutdown Panel "C" (TDAFW panel) and the preferred method of communications to co-ordinate activities with personnel outside the Control Room. Knowing where to report and how to communicate meets the KA.

EXPLANATION OF REQUIRED KNOWLEDGE

Per AOP 18038-1 step 7, the System Operator goes to Shutdown Panel C (TDAFW Pump Room). The Unit Operator goes to Shutdown Panel B, unless he is the ENN Communicator, in which case he would go to the TSC.

Per AOP 18038-1 step 14, communication is to be established between all stations, preferably using sound powered telephone remote shutdown channel, red box. The RNO of this step is to utilize either Bridge Phone Ext 3145, Page, or Radio.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The System Operator is to report to Shutdown Panel C (TDAFW Pump Room). However, the report position of the UO is not commonly asked. It is reasonable for a candidate to believe that the UO would report to Shutdown Panel C since a System Operator has little experience and training in controlling SG levels, and this is normally the responsibility of the UO in the Main Control Room.

The second part is correct. Per AOP 18038-1 step 14, communication is to be established between all stations preferably using sound powered telephone remote shutdown channel, red box.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per AOP 18038-1 step 14, communication is to be established between all stations preferably using sound powered telephone remote shutdown channel, red box. However, the RNO of this step is to utilize either Bridge Phone Ext 3145, Page, or Radio. It is reasonable for a candidate without specific knowledge of the procedural requirement to conclude that the bridge phone would be a "better" option due to sound quality issues and need for headsets.

C. Correct. The first part is correct. Per 18038-1 step 7, the System Operator is to report to Shutdown Panel C (TDAFW Pump Room).

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.1.8	
Importance Rating:	3.4 / 4.1	
Technical Reference:	18038-1, Rev 33.7, pages 10 & 15	
References provided:	None	
Learning Objective:	LO-PP-60327-13	State the locations where the following operators will be stationed during operation from Remote Shutdown Panels. b. Reactor Operator (OATC) g. Extra shift personnel
	LO-TA-20016	Start the TDAFW Pump from Shutdown Panel "C" using 18038-1/2
	LO-TA-60047	Establish control from remote shutdown panels
Question origin:	BANK - Direct Reuse of HL17 NRC G2.1.8	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.10 / 45.5 / 45.12 / 45.13	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. NOTE

ATTACHMENT H is to be used at SS discretion as an aid for operators dispatched to perform local actions.

CAUTION

Fire event qualified instrumentation is only available on Shutdown Panel B and marked in red.

__7. Make a page announcement that the Control Room is being evacuated and perform the following:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Dispatch Operators to the following locations:	a. IF insufficient personnel are available, THEN use the following priority:
1) Shutdown Panel B (CB-A43): __• Shift Supervisor __• Extra Shift Personnel	1) Shutdown Panel B.
2) Shutdown Panel A (CB-A75): __• Reactor Operator	2) Shutdown Panel A.
3) Shutdown Panel C (TDAFW Pump Room): __• System Operator	3) Shutdown Panel C (TDAFW Pump Room).
	4) TSC - Plant Computer Terminal.

Step 7 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__14. Establish communications between all stations (preferably sound powered telephones remote shutdown channel, red box).	__14. All stations on same channel or line: __Bridge Phone Ext 3145, codes 123# or 234#. __Page __Radio
__15. Initiate the Continuous Actions Page.	
__*16. **Monitor for Control Room habitability.**	__*16. WHEN Control Room conditions become habitable, THEN go to Step 76 to continue recovery actions.
__17. Place all transfer switches on 1AA02-00 (CB-A48) to LOCAL.	
__18. Place all transfer switches on 1BA03-00 (CB-A50) to LOCAL.	
*19. **Check at least one ACCW Pump RUNNING (approximately 62 amps):** __ 1AA02-15 -OR- __1BA03-20	*19. Perform the following: __a. Stop all RCPs. __b. Isolate letdown by closing LETDOWN ISOLATION VLV UPSTREAM 1-LV-460 (Shutdown Panel A) and LETDOWN ISOLATION VLV DOWNSTREAM 1-LV-459 (Shutdown Panel A.)

1. G2.2.15 001/LOIT AND LOCT/RO/C/A 3.9/4.3/G2.2.15/LO-TA-63007///

Initial conditions:

- Unit 1 is at 45% reactor power.
- EHC Pump ‘A’ discharge filter #8 is in service.
- EHC Pump ‘A’ discharge filter #4 is tagged out for replacement.

Current conditions:

- The following EHC Pump ‘A’ discharge filter #4 valve tags are ready for release:

  1-1615-U4-592, EHC HYD PUMP ‘A’ DISCH FILTER #4 INLET ISO
  1-1615-U4-593, EHC HYD PUMP ‘A’ DISCH FILTER #4 OUTLET ISO

- EHC Pump ‘A’ discharge filter #8 is to remain in service.

Which one of the following completes the following statement?

Both valves will be __(1)__ after the tagout is released to standby alignment,

and

if EHC pressure were to lower to 1000 psig, the reactor __(2)__ trip.

REFERENCE PROVIDED

	__(1)__	__(2)__
A.	open	would
B.	open	would NOT
C.✓	closed	would
D.	closed	would NOT

K/A

G2.2.15 **Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.**

K/A MATCH ANALYSIS

The KA addresses the relationship between plant documents and component status control to include asking the candidate to determine how to align components based on plant conditions. In addition, the candidate must determine which condition would cause a main turbine trip if EHC pressure lowered during the clearance release.

EXPLANATION OF REQUIRED KNOWLEDGE

Per lineup 11840-1, both valves have a required position of OPEN with a note that modifies the required position if filter 8 is in service, stating that the valve should be CLOSED in that case. Additionally, P&ID 1X4DB194 shows both valves as normally open. Both of these documents are given as references. Candidates are then required to reconcile the contradiction between the documents. P&IDs are utilized during tagout release preparation, but the lineup takes priority. All discrepancies must be carefully scrutinized, and compared to the operator's knowledge of system design and current plant configuration.

Note: There have been several instances in plant OE of inexperienced licensed operators releasing tagouts based on P&ID position alone. They did not reference the lineups or resolve discepancies between the two, reslulting in misposition events.

With EHC pressure lowered to less than 1100 psig, the turbine will trip. Since reactor power is above the P-9 setpoint of 40%, the reactor will also trip.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per lineup 11840-1, these valves should be released CLOSED since filter 8 is in service. However, a candidate may believe the P&ID takes priority, or fail to reference the lineup and conclude the required position is OPEN.

The second part is correct. Per ALB20-D01, with EHC pressure lowered to less than 1100 psig, the turbine will trip. Since reactor power is above the P-9 setpoint of 40%, the reactor will also trip.

B.Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. Per ALB20-D01, with EHC pressure lowered to less than 1100 psig, the turbine will trip. Since reactor power is above the P-9 setpoint of 40%, the reactor will also trip. However, a candidate can easily confuse the P-8 setpoint of 48% with the P-9 setpoint of 40% and assume the turbine will trip and the reactor will not. Or, a candidate could confuse the alarm setpoint of 1500 psig and the EHC pump autostart setpoint of 1400 psig with the turbine trip setpoint of 1100 psig and believe that the turbine will not trip.

C. Correct. The first part is correct. Per lineup 11840-1, these valves should be released CLOSED since filter 8 is in service.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO
Tier # / Group #	T3
K/A#	G2.2.15
Importance Rating:	3.9 / 4.3

Technical Reference: LINEUP 11840-1 "EHC System Alignment' page 4, Rev 14.1
ARP 17020-1, Rev 53.2, pages 41, 42, & 68
SOP 13503A-1, Rev 7.2, page 33
P&ID 1X4DB194, Rev 29.0

References provided: LINEUP 11840-1 "EHC System Alignment' page 4, Rev 14.1
P&ID 1X4DB194, Rev 29.0

Learning Objective:
LO-PP-30103-04 Describe EHC pumps normal operating pressure.
LO-PP-30201-12 State the signals that will generate a turbine trip; excluding specific generator trips.
LO-TA-63007 Tagout Review in accordance with NMP-AD-003

Question origin: NEW

Cognitive Level: C/A

10 CFR Part 55 Content: 41.10 / 43.3 / 45.13

Comments: Question appears to match the KA.
All of question (2) is LOD=1 - I would expect that all operators should already know the appropriate way to perform a check of a valve, whether it is open or closed.
Question (1) is a LOD=1 - the answer for part (1) is a direct lookup from the provided lineup, thereby making all of the other distractors non plausible.
Recommend replacing this question. Need to make sure that the selected reference doesn't lend itself to a direct lookup - i.e., you could use a reference for one system where the way another system is aligned impacts the correct answer for the given system. The applicant would need to have plant specific knowledge of the secondary system to know which answer is correct.
-JAT 12/19/2013

Question is improved, but need to ensure there is only one correct answer. Is "standby alignment" defined anywhere? Is it possible that someone could successfully argue that

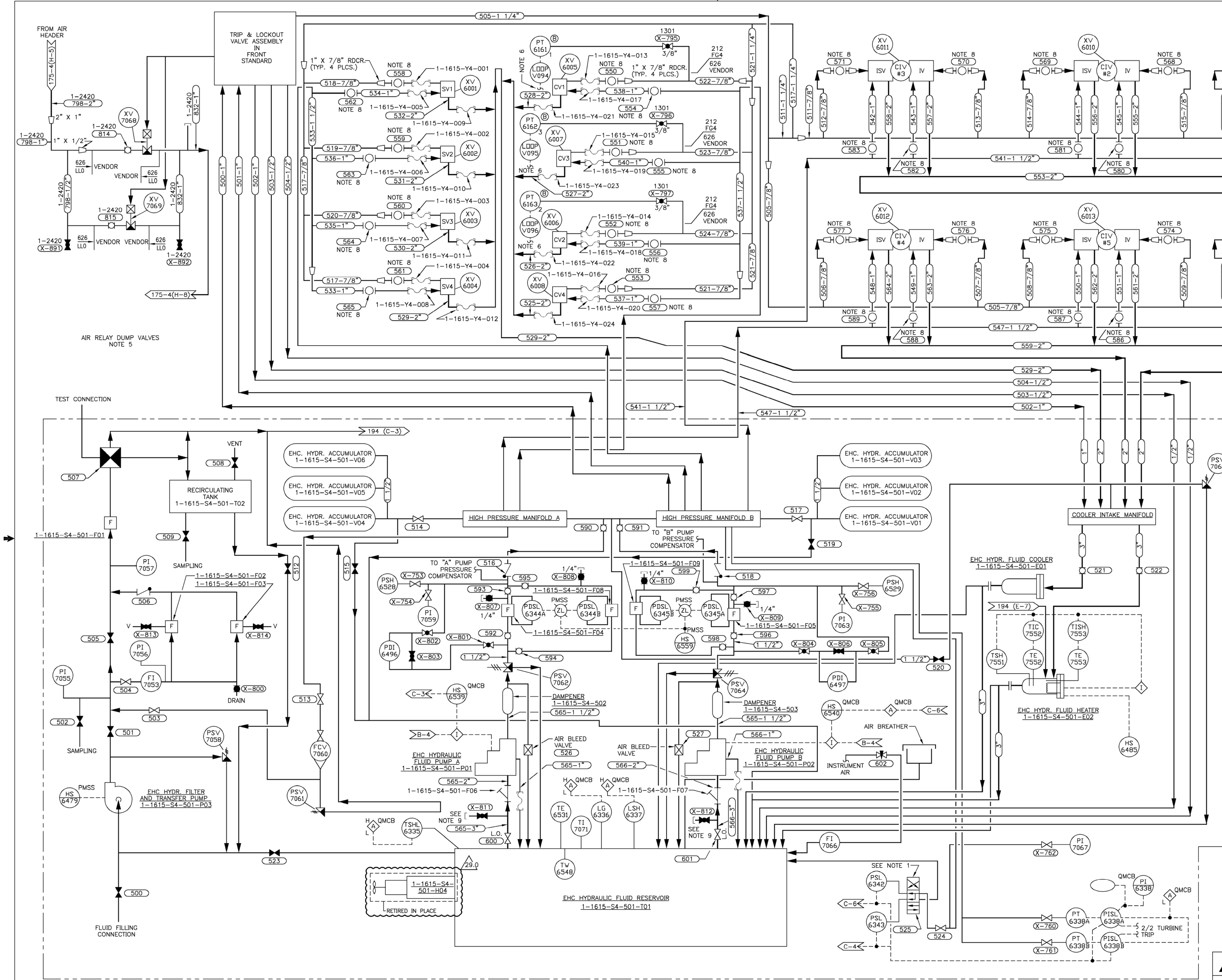
returning Filter #4 to service and taking filter #8 out of service meets "standby alignment" (thereby possibly having two correct answers to this question)?
- JAT 2/4/14

A bullet was added to current conditions that states "EHC Pump ‘A’ discharge filter #8 is to remain in service" to eliminate the possible contention basis.
- JCC 4/3/14

You have completed the test!

COMPONENT	DESCRIPTION	CONDITION REQUIRED	LINEUP (INITIALS)
LOCATED AT THE HYDRAULIC POWER UNIT - TURBINE BLDG LEVEL 1			
1-1615-U4-590	EHC HYD PUMP A DISCHARGE ISOLATION	OPEN	________
1-1615-U4-592	EHC HYD PUMP A DISCH FILTER #4 INLET ISO	OPEN (1)	________
1-1615-U4-593	EHC HYD PUMP A DISCH FILTER #4 OUTLET ISO	OPEN (1)	________
1-1615-U4-594	EHC HYD PUMP A DISCH FILTER #8 INLET ISO	CLOSED (2)	________
1-1615-U4-595	EHC HYD PUMP A DISCH FILTER #8 OUTLET ISO	CLOSED (2)	________
1-1615-X4-807	EHC HYD PUMP A DISCH FILTER #4 VENT	CLOSED	________
1-1615-X4-808	EHC HYD PUMP A DISCH FILTER #8 VENT	CLOSED	________
1-1615-X4-801	EHC HYD PUMP A DISCH FILTER PDI-6496 HI SIDE ISOLATION	OPEN	________
1-1615-X4-802	EHC HYD PUMP A DISCH FILTER PDI-6496 LOW SIDE ISOLATION	OPEN	________
1-1615-X4-803	EHC HYD PUMP A DISCH FILTER PDI-6496 BYPASS	CLOSED	________
1-1615-U4-591	EHC HYD PUMP B DISCHARGE ISOLATION	OPEN	________
1-1615-U4-596	EHC HYD PUMP B DISCH FILTER #5 INLET ISO	OPEN (3)	________
1-1615-U4-597	EHC HYD PUMP B DISCH FILTER #5 OUTLET ISO	OPEN (3)	________
1-1615-U4-598	EHC HYD PUMP B DISCH FILTER #9 INLET ISO	CLOSED (4)	________

(1) CLOSED IF FILTER 8 SELECTED FOR SERVICE
(2) OPEN IF FILTER 8 SELECTED FOR SERVICE
(3) CLOSED IF FILTER 9 SELECTED FOR SERVICE
(4) OPEN IF FILTER 9 SELECTED FOR SERVICE



EHC HYDR. POWER UNIT
1-1615-S4-501

NOTES:

1. THREE WAY TEST VALVE FOR TESTING FUNCTION OF PSL6342 & PSL6343.
2. ALL PIPING, INSTRUMENTATION, & EQUIPMENT VENDOR SUPPLIED EXCEPT AS NOTED.
3. DELETED.
4. DELETED.
5. AIR RELAY DUMP VALVES ARE IN SYSTEM 1615.
6. REFERENCE REACTOR TRIP LOGIC 1X5DN203-1. PRESSURE TRANSMITTER IS A COMPONENT OF MAIN STEAM SYSTEM 1301.
7. ALL 1615 SYSTEM PIPING, VALVES AND EQUIPMENT ARE PROJECT CLASS 626 UNLESS OTHERWISE SHOWN.
8. VALVE IS EQUIPPED WITH A LOCKING DEVICE AND IS NORMALLY LOCKED OPEN TO ENSURE TURBINE OVERSPEED PROTECTION AVAILABILITY (REF.: DCP 90-V1N0096-0-1).
9. TEMPORARY FILTRATION EQUIPMENT MAY BE CONNECTED TO TAKE SUCTION FROM VALVE 1-1615-X4-811 OR 812 AND RETURN THE FILTERED FLUID TO A SPARE PENETRATION OF THE RESERVOIR. REFERENCE RER 2003-V0375.

CAD 1X4DB194
Acad2004 TJS - 02

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
29.0	REVISED PER ABN- V00852, VER. 1.0.	8-07-07	TJS	MWD	WLN	X

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
ELECTRO-HYDRAULIC
CONTROL (EMC) SYSTEM
SYSTEM NO. 1615

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB194	29.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

ATTACHMENT C Sheet 1 of 6

PERMISSIVES, CONTROL INTERLOCKS, REACTOR TRIPS AND ESF ACTUATIONS

PERMISSIVES

Permissive	Setpoint/Coincidence	Function
P-4	Train related Rx trip & Bypass breaker open	Trips Main Turbine Train A - mechanical Train B - electrical FWI if Lo Tavg (2/4 ≤ 564 °F) present Seals in FWI if caused by SI or P-14 (Hi Hi Level) Must be present to block auto SI after SI reset. P-4 Train A arms Steam Dumps P-4 Train B swaps Steam Dumps to plant trip controller
P-6	1/2 IR Detectors ≥ 2.0 E –5 % Rx Power	Allows manual block of SR High ϕ trip
P-7	P-10 (2/4 PR NIs ≥ 10% Rx power) or P-13 (PT-505 or 506 ≥ 10% turbine power)	Unblocks "At Power" Trips Przr Low Pressure Przr High Level RCS Two Loop Low Flow RCP UF RCP UV
P-8	2/4 PR NIS ≥ 48% Rx power	Enables Single Loop Low Flow Rx Trip
P-9	2/4 PR NIS ≥ 40% Rx power	Enables Turbine trip Rx trip
P-10	2/4 PR NIS ≥ 10% Rx power	Auto block of SR High ϕ trip Enables P-7 Allows manual block of IR rod stop and Hi ϕ trip Allows manual block of PR Hi ϕ trip Lo Setpoint
P-11	2/3 Przr Pressure channels ≤ 2000 psig	Auto enables Lo Przr Press SI & Lo Steamline Press SI/SLI & sends signal to open Accum Isolation Valves when P-11 resets. P-11 allows operator to block PRZR & Steamline low pressure SI & SLI. Also activates "Not Full Open" annunciators for Accumulator MOVs (ALB16; A5, B5, C5 & D5) and HV-8806; (ALB16 E03).
P-12	2/4 NR Tavg ≤ 550 °F	Interlocks Steam Dumps closed (Cooldown Dump Valves PV-507A, B & C) may be reopened by use of Bypass Interlock switches)
P-13	1/2 Turbine Impulse channels ≥ 10%	Enables P-7
P-14	2/4 NR SG Level channels ≥ 82%	Actuates FWI Actuates MFP and Main Turbine trip

WINDOW D05

ORIGIN	SETPOINT	
1-PISL-6338A 1-PISL-6338B	1500 psig	HYD FLUID LO PRESS

1.0 PROBABLE CAUSE

1. Failure of Electrohydraulic Control (EHC) Fluid Pumps.
2. Clogged strainers and filters in pump suction or discharge.
3. EHC Fluid System leak.

2.0 AUTOMATIC ACTIONS

1. If pressure drops below 1400 psig, the standby EHC Fluid Pump will start.
2. If pressure continues to drop to 1100 psig, the Turbine will trip.

3.0 INITIAL OPERATOR ACTIONS

1. IF a reactor trip occurs, **Go To** 19000-C, "E-0 Reactor Trip Or Safety Injection."
2. **Verify** standby EHC Fluid Pump is on, if needed.

4.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

EHC fluid is a fire resistant fluid that may be harmful to personnel. Observe proper safety precautions when in contact with this fluid.

1. **Dispatch** an operator to the Hydraulic Power Unit to **check** for system leaks or pump failure.
2. IF equipment failure is indicated, **initiate** maintenance as required.

WINDOW D05
(Continued)

5.0 **COMPENSATORY OPERATOR ACTIONS**

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB194, 1X3D-BC-Q56B, 1X4AA01-280-0, CX5DT1101-17A

WINDOW E05

ORIGIN	SETPOINT	
2 out of 2: 1-PISL-6338A 1-PISL-6338B	1100 psig with a 3-sec time delay	HYD FLUID LO PRESS TURB TRIP

1.0 **PROBABLE CAUSE**

1. Failure of Electrohydraulic Control (EHC) Fluid Pumps.

2. Clogged strainers and filters in pump suction or discharge.

3. EHC Fluid System leak.

2.0 **AUTOMATIC ACTIONS**

Turbine trip.

3.0 **INITIAL OPERATOR ACTIONS**

1. IF a reactor trip has occurred, **Go To** 19000-C, "E-0 Reactor Trip Or Safety Injection."

2. IF a reactor trip has NOT occurred, **Go To** 18011-C, "Turbine Trip Below P-9."

4.0 **SUBSEQUENT OPERATOR ACTIONS**

NONE

5.0 **COMPENSATORY OPERATOR ACTIONS**

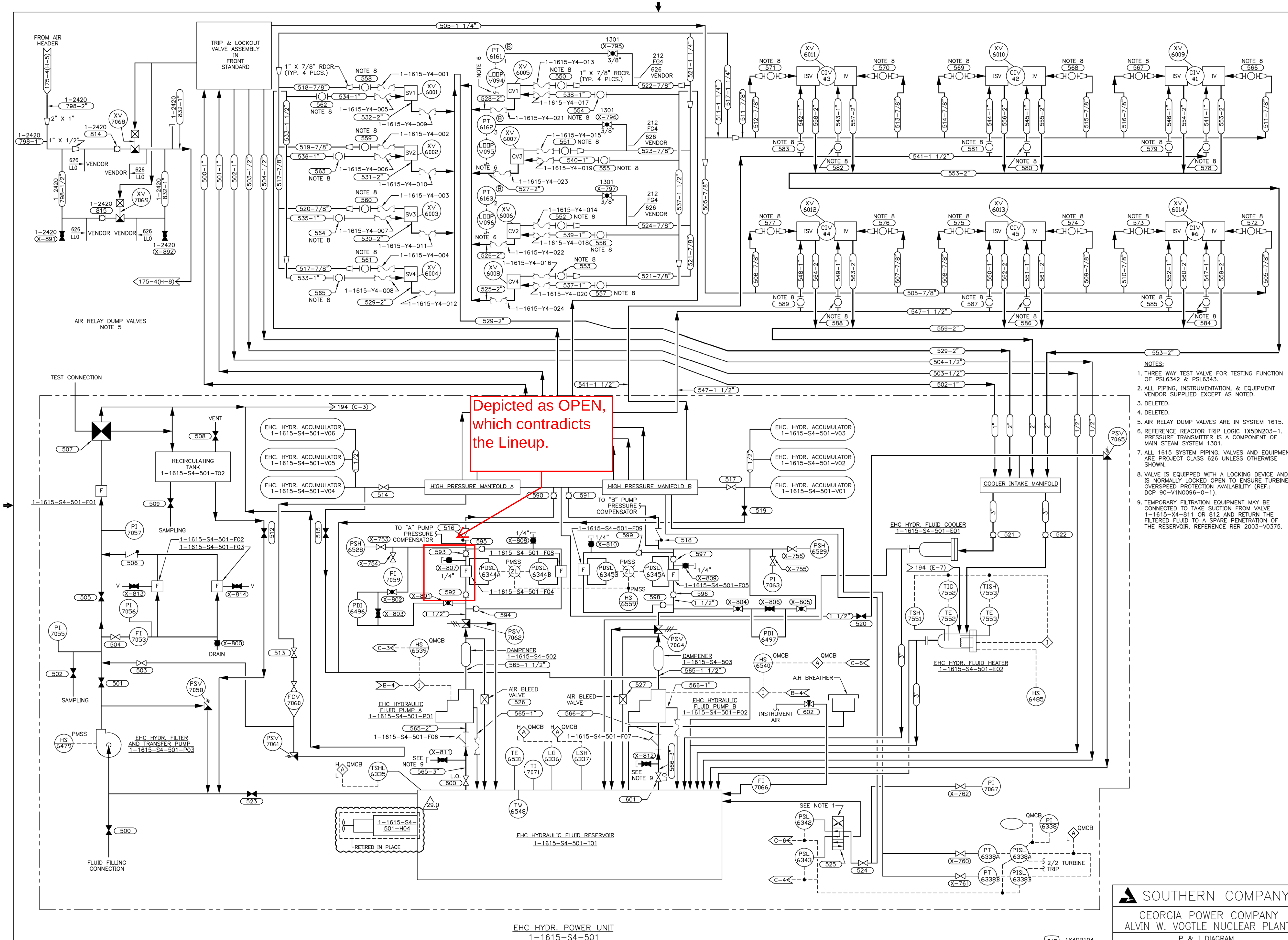
NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB194, 1X3D-BC-Q56B, 1X4AA01-280-0, 1X5DN203-1, CX5DT1101-17A

COMPONENT	DESCRIPTION	CONDITION REQUIRED	LINEUP (INITIALS)
LOCATED AT THE HYDRAULIC POWER UNIT - TURBINE BLDG LEVEL 1			
1-1615-U4-590	EHC HYD PUMP A DISCHARGE ISOLATION	OPEN	________
1-1615-U4-592	EHC HYD PUMP A DISCH FILTER #4 INLET ISO	OPEN (1)	________
1-1615-U4-593	EHC HYD PUMP A DISCH FILTER #4 OUTLET ISO	OPEN (1)	________
1-1615-U4-594	EHC HYD PUMP A DISCH FILTER #8 INLET ISO	CLOSED (2)	________
1-1615-U4-595	EHC HYD PUMP A DISCH FILTER #8 OUTLET ISO	CLOSED (2)	________
1-1615-X4-807	EHC HYD PUMP A DISCH FILTER #4 VENT	CLOSED	________
1-1615-X4-808	EHC HYD PUMP A DISCH FILTER #8 VENT	CLOSED	________
1-1615-X4-801	EHC HYD PUMP A DISCH FILTER PDI-6496 HI SIDE ISOLATION	OPEN	________
1-1615-X4-802	EHC HYD PUMP A DISCH FILTER PDI-6496 LOW SIDE ISOLATION	OPEN	________
1-1615-X4-803	EHC HYD PUMP A DISCH FILTER PDI-6496 BYPASS	CLOSED	________
1-1615-U4-591	EHC HYD PUMP B DISCHARGE ISOLATION	OPEN	________
1-1615-U4-596	EHC HYD PUMP B DISCH FILTER #5 INLET ISO	OPEN (3)	________
1-1615-U4-597	EHC HYD PUMP B DISCH FILTER #5 OUTLET ISO	OPEN (3)	________
1-1615-U4-598	EHC HYD PUMP B DISCH FILTER #9 INLET ISO	CLOSED (4)	________

(1) CLOSED IF FILTER 8 SELECTED FOR SERVICE
(2) OPEN IF FILTER 8 SELECTED FOR SERVICE
(3) CLOSED IF FILTER 9 SELECTED FOR SERVICE
(4) OPEN IF FILTER 9 SELECTED FOR SERVICE



NOTES:

1. THREE WAY TEST VALVE FOR TESTING FUNCTION OF PSL6342 & PSL6343.
2. ALL PIPING, INSTRUMENTATION, & EQUIPMENT VENDOR SUPPLIED EXCEPT AS NOTED.
3. DELETED.
4. DELETED.
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8. VALVE IS EQUIPPED WITH A LOCKING DEVICE AND IS NORMALLY LOCKED OPEN TO ENSURE TURBINE OVERSPEED PROTECTION AVAILABILITY (REF.: DCP 90-V1N0096-0-1).
9. TEMPORARY FILTRATION EQUIPMENT MAY BE CONNECTED TO TAKE SUCTION FROM VALVE 1-1615-X4-811 OR 812 AND RETURN THE FILTERED FLUID TO A SPARE PENETRATION OF THE RESERVOIR. REFERENCE RER 2003-V0375.

CAD 1X4DB194
Acad2004 TJS - 02

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
ELECTRO-HYDRAULIC
CONTROL (EMC) SYSTEM
SYSTEM NO. 1615

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
29.0	REVISED PER ABN- V00852, VER. 1.0.	8-07-07	TJS	MWD	WLN	X

SCALE: NONE	DRAWING NO.	VER.
JOB NO.10604	1X4DB194	29.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL

1. G2.2.23 001/LOIT/RO/C/A 3.1/4.6/G2.2.23/LO-LP-61202-06///

Procedure title as follows:

- 14807A-1, "Train 'A' Motor Driven Auxiliary Feedwater Pump and Check Valve Inservice and Response Time Test"

At time 1000:

- Unit 1 is at 7% reactor power with a startup in progress.
- MDAFW pump 'A' discharge valves, 1HV-5137 and 1HV-5139, to the SGs are closed by the UO to perform a surveillance per 14807A-1.

At time 1005:

- MDAFW pump 'A' is started.

At time 1010:

- MDAFW pump 'A' is stopped.

At time 1015:

- Train 'A' MDAFW system is returned to normal standby alignment.

Which one of the following completes the following statement?

During performance of the surveillance and per Tech Spec 3.7.5, "Auxiliary Feedwater (AFW) System," the UO would track an LCO not met (out-of-service) time of __(1)__ minutes,

and

if a Train 'A' MDAFW actuation signal were received during the surveillance when the discharge valves were closed, the valves would __(2)__.

	__(1)__	__(2)__
A.	5	automatically open
B.	5	remain closed
C.✓	15	automatically open
D.	15	remain closed

K/A

G2.2.23 Ability to track Technical Specification limiting conditions for operations.

K/A MATCH ANALYSIS

The question tests the candidate's ability to track Technical Specifications, and, using procedure guidance, to correctly track safety related equipment OOS time associated with an AFW surveillance.

EXPLANATION OF REQUIRED KNOWLEDGE

Per OSP 14807A-1 CAUTION prior to step 5.1.3, the Train A MDAFW pump is inoperable as soon as either discharge valve is closed. Therefore, the LCO would not be met starting at time 10:00 and would remain not met until 10:15 when the AFW system was restored to standby alignment. The logged out-of-service time would be 15 minutes.

Per ELEMENTARY 1X3D-BC-F08C, the discharge valve handswitches are spring return to auto. Once the discharge valves were stroked closed, the handswitch would be in AUTO. Therefore, when the AFW Actuation signal was processed, both discharge valves would stroke fully open. Even though AFW discharge valves will automatically stroke open, per commitments 1984301714 and 1984301715, AFW will be considered inoperable with discharge valves closed and reactor power >5% RTP.

NOTE: The first part of the question is associated with AFW OPERABILITY. It is part of the normal RO job function to notify the SS when an LCO is not met as part of a surveillance as well as when the inoperable condition is restored. The inoperability call is explicitely called out in the surveillance in a CAUTION. The OPERABILITY requirement associated with AFW discharge valves is taught as part of systems training. Therefore, this is an RO level function.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. Per OSP 14807A-1 CAUTION prior to step 5.1.3, the Train A MDAFW pump is inoperable as soon as either discharge valve is closed. Therefore, the LCO would not be met starting at time 10:00 and would remain not met until 10:15, when the AFW system was restored to standby alignment. The logged out-of-service time would be 15 minutes. However, the candidate may determine that since the discharge valves would stroke open on an automatic signal, the LCO time would only be associated with the time from when the AFW pump was stopped until standby alignment was achieved, or from when the discharge valves were closed until the pumps were started. Either of these would be recorded as 5 minute durations.

The second part is correct. The discharge valve handswitches

are spring return to auto. Once the discharge valves were stroked closed, the handswitch would be in AUTO. Therefore, if an AFW Actuation signal was received, both discharge valves would stroke fully open.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. The discharge valve handswitches are spring return to auto. Once the discharge valves were stroked closed, the handswitch would be in AUTO. Therefore, if an AFW Actuation signal was received, both discharge valves would stroke fully open. However, the candidate may determine that since the required alignment is open, the valves do not receive an automatic signal. In addition, the knowledge that the open signal can be overridden and the valves not automatically open makes this selection plausible.

C. Correct. The first part is correct. Per OSP 14807A-1 CAUTION prior to step 5.1.3, the Train A MDAFW pump is inoperable as soon as either discharge valve is closed. Therefore, the LCO would not be met starting at time 10:00 and would remain not met until 10:15 when the AFW system was restored to standby alignment. The logged out-of-service time would be 15 minutes.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

You have completed the test!

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.2.23	
Importance Rating:	3.1 / 4.6	
Technical Reference:	UOP 12004-C, Rev 108.0, page 25 SOP 13610-1, Rev 50.4, page 10 & 96 OSP 14870A-1, Rev 5.0, page 10 LESSON PLAN V-LO-PP-20101, Rev 3.2, slide 14 ELEMENTARY 1X3D-BC-F08C, Rev 8.0	
References provided:	None	
Learning Objective:	LO-LP-61202-06	Describe the basic steps required to place the AFW System in standby readiness.
	LO-PP-20101-01	Describe when the AFW system is used to support normal power operations and what the maximum power AFW operation is used for.
	LO-PP-20101-02	Describe the normal at power standby alignment of the AFW system.
	LO-PP-20101-04	List the AFW system automatic start signals and component actuations.
	LO-LP-63404-02	Describe the following as applicable to the surveillance test program: a. purpose of surveillance work orders b. where the procedure number to be used for performance of surveillance tests are identified c. who must authorize the performance of tests that manipulate or affect plant equipment d. who reviews the test results to confirm that they satisfy the acceptance criteria e. purpose of surveillance test f. failure of surveillance tests g.duties and responsibilities of surveillance test performer if a test fails or cannot be completed within the specified time
	LO-TA-61010	Perform Power Ascent During Low Power Operations using 12004-C
Question origin:	NEW	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.4 / 41.7 / 41.10	

Comments:

You have completed the test!

INITIALS

2.2 LIMITATIONS

2.2.1 Technical Specification LCO 3.7.5 requires that three independent Auxiliary Feedwater trains be operable in MODES 1, 2, or 3. TS SR 3.7.5.1 requires that each automatic valve in the discharge flowpath must be in the fully open position for standby readiness. _________

Testing of TDAFW Pump in accordance with Technical Specification SR 3.7.5.2 is not required to be performed until 24 hours after steam generator pressure is greater than or equal to 900 psig. _________

2.2.2 The TDAFW Pump Turbine must be coupled to the pump prior to entry into MODE 3 per Technical Specification LCO 3.7.5. _________

2.2.3 Technical Specification LCO 3.7.6 requires that one Condensate Storage Tank (CST) be operable, with a safety-related volume of greater than or equal to 340,000 gallons (66% of span) when in Modes 1, 2, or 3.
(SNC16060, 1996332946, (SNC16059, 1996332945) _________

2.2.4 To prevent pump degradation and abnormal wear, the following minimum flow requirements have been established for the MDAFW and TDAFW Pumps:

a. For short periods of operation not to exceed three (3) hours in a 24-hour period:; (SNC15010, 1990319699)

(1) MDAFW Pump greater than or equal to 150 gpm. _________

(2) TDAFW Pump greater than or equal to 53 gpm @ 1535 rpm varying linearly to 145 gpm @ 4230 rpm (see Figure 1). _________

b. For continuous pump operation greater than three (3) hours:

(1) MDAFW Pump greater than or equal to 265 gpm. _________

(2) TDAFW Pump greater than or equal to 175 gpm @ 1535 rpm varying linearly to 450 gpm @ 4230 rpm (see Figure 2 _________

CHECKLIST 2 - AUXILIARY FEEDWATER SYSTEM ALIGNMENT FOR STANDBY READINESS

Sheet 1 of 3

NOTES

- This checklist should be performed on non-operating equipment only. Any equipment which is currently in operation should be marked N/A and noted in the comment section of this checklist. ☐
- The AFW System is OPERABLE in MODES 1-3 with the MDAFW Pump Miniflow Isolation Valves (1-1302-U4-769 and 770) either open or closed. The preferred position for standby readiness is closed because the system is more tolerant of potential failure of 1-FV-5154 and 5155 (and associated controls) when 1-1302-U4-769 and 770 are closed. ☐
- Any of the following sections in this checklist should be marked N/A if not used. ☐

I. Motor Driven Auxiliary Feedwater Pump A Alignment:

	COMPONENT	DESCRIPTION	CONDITION REQUIRED	LINEUP (INITIALS)	VERIFICATION (INITIALS)
(1)	1HS-5139A	SG-1 FROM MDAFW PMP-A	OPEN	________	________
(2)	1HS-5137A	SG-4 FROM MDAFW PMP-A	OPEN	________	________
(3)	1HS-5131A	MDAFW-A 1-1302-P4-003	AUTO	________	________
(4)	OVERCURRENT LOR MDAFW-A 1-1302-P4-003 (At breaker 1AA02-17)		RESET	________	________
(5)	FV-5155	AFW P-3 MINI FLOW (ZLB-4 INDICATION LIT)	OPEN	________	________
(6)	1-1302-U4-769	AFW MDAFW PUMP A MINI-FLOW ISO VALVE	CLOSED	________	________
(7)	MDAFW PUMP A suction & discharge piping .		NOT hot to touch	________	

INITIALS

o. Outage management has reviewed the outage schedule and determined that no impacts to mode change exist. _________

Outage Management Representative

____
(Signature)

p. Engineering review of 00309-C, Control of Unattended Temporary Material In Containment In Modes 1-4 has been completed for Mode 1 entry and any restrictions are known and resolved. _________

q. **Notify** SM that conditions are met to change status from Mode 2 to Mode 1. (SM to document authorization to change modes on Checklist 1.) _________

4.1.16 PRIOR to exceeding 5% reactor power, as read on the highest reading PR NIS or highest reading Loop ΔT, **verify** the AFW has been aligned for Standby per Checklist 2 of 13610, “Auxiliary Feed Water System." (1984301714, 1984301715) _________*

NOTE

The purpose of this step is to ensure no obstructions are present in steam lines or feed lines. ☐

4.1.17 **Perform** a channel check of NIS and Delta-Ts and **initiate** periodic monitoring to ensure all four loops track power increase together. _________

INITIALS

5.1 TEST OF MDAFW PUMP A

5.1.1 **Check** Train A MDAFW Pump suction pressure on 1-PI-5129A is 15 psig or greater and record pressure.

________psig ________

5.1.2 IF performing response time test,

a. **Calculate** the pressure at which the stop watch will be stopped for measuring response time to discharge pressure. ________

Discharge Pressure = Suction Pressure + Minimum ΔP

Discharge Pressure = ___________ + 1625 psid

Discharge Pressure = ___________ psig

b. **Record** target discharge pressure in Step 5.1.11.2 a. ________

> **CAUTION**
>
> The following step will render Train A MDAFW Pump INOPERABLE. (Technical Specifications LCO 3.7.5) ☐

5.1.3 **Close** the following valves:

Critical

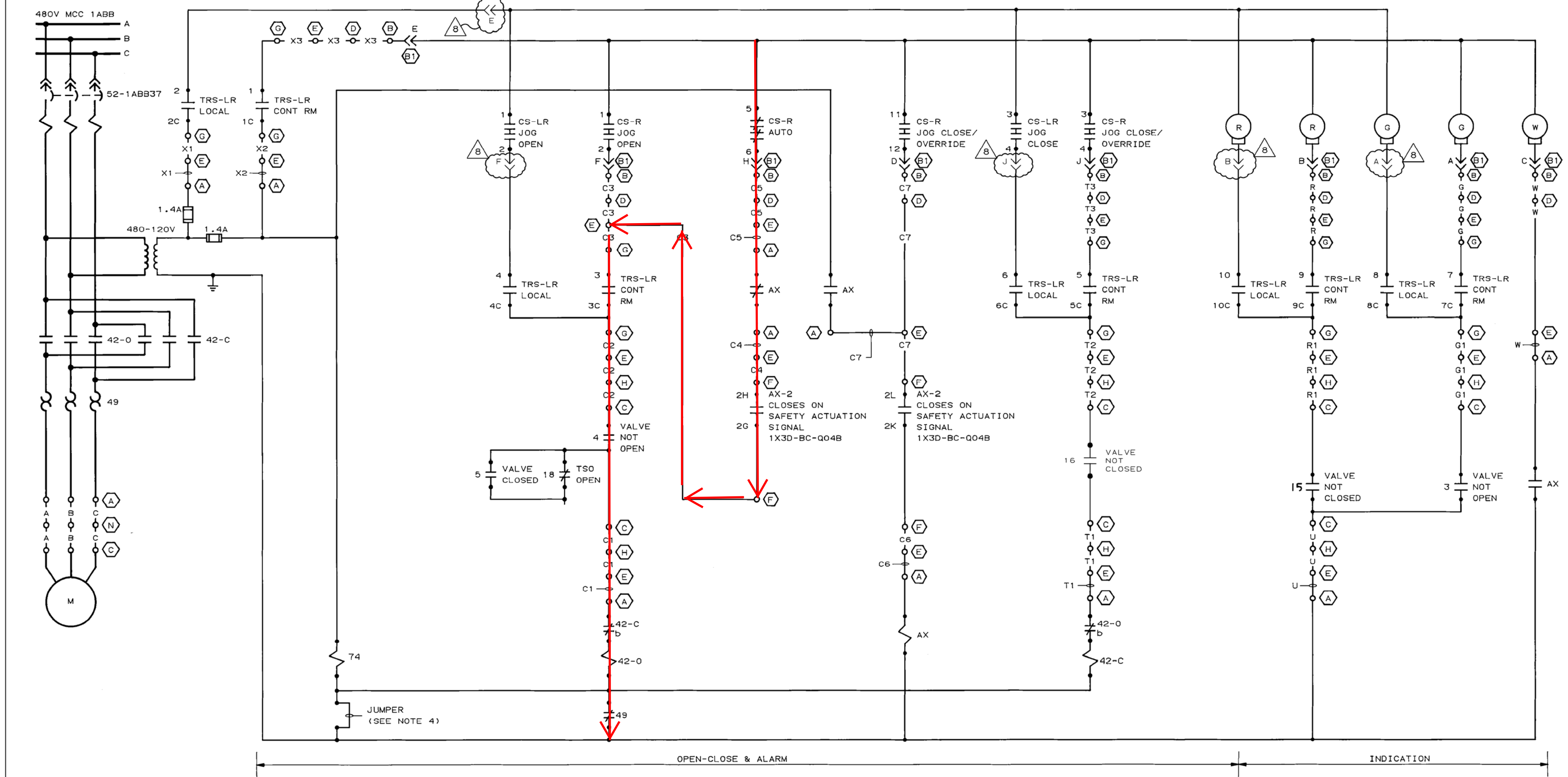
- SG-4 FROM MDAFW PMP-A 1-HV-5137 ________

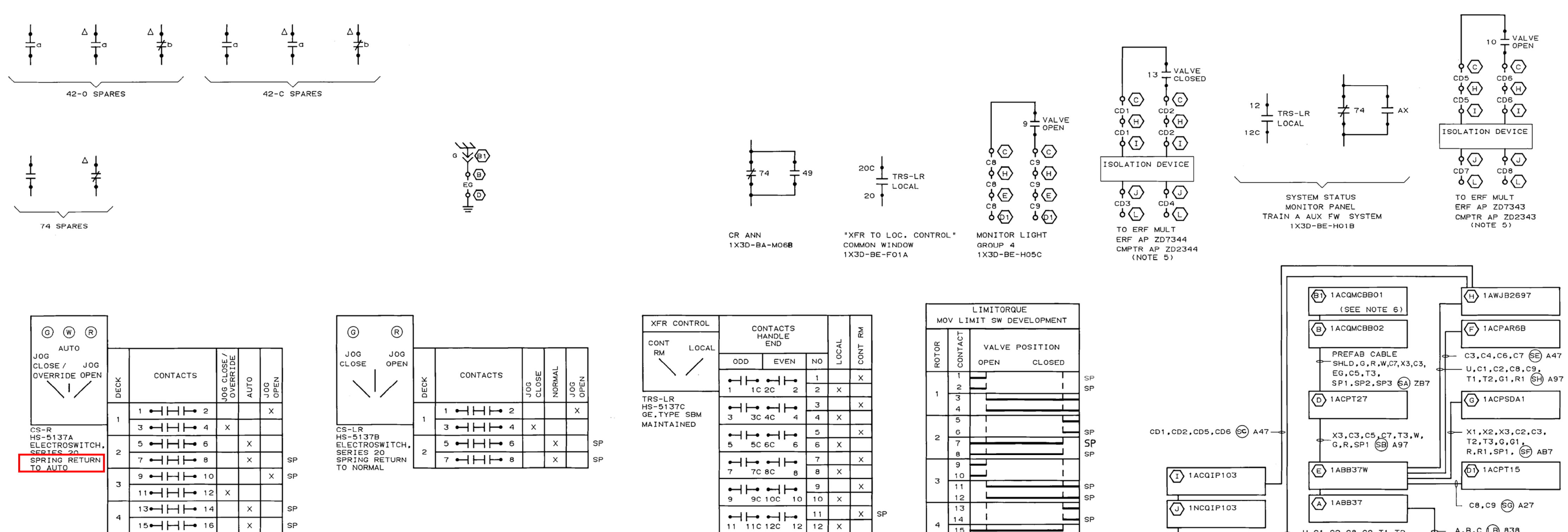
________ CV

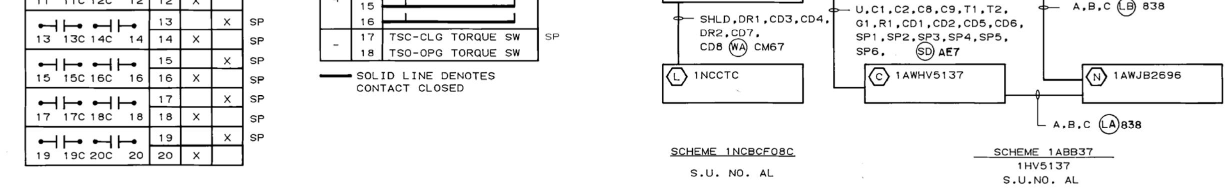
Critical

- SG-1 FROM MDAFW PMP-A 1-HV-5139 ________

________ CV







NOTES:

1. FOR GENERAL NOTES AND SYMBOLS REFER TO DRAWINGS LISTED ON 1X3D-AA-A00B.
2. ⬡ DENOTES EQUIPMENT LOCATION NUMBER, FOR COMPLETE EQUIPMENT LOCATION NUMBER REFER TO CABLING BLOCK DIAGRAM.
3. SPARE CONTACT WITH 'Δ' IS WIRED TO UNIT TERMINAL BLOCKS.
4. FIELD TO REMOVE JUMPER FROM THERMAL OVERLOAD CONTACT DURING INITIAL START-UP & PERIODIC TESTING.
5. CIRCUITS AFTER THE ISOLATION DEVICE ARE CONSIDERED NON-CLASS 1E .
6. HS-5137A IS LOCATED AT B1 INTERNAL CONNECTION BETWEEN B1 AND B IS BY WESTINGHOUSE THE PREFAB CABLE CONNECTOR IS LOCATED AT B .
7. ROTOR NUMBER 2 SHALL BE USED ONLY FOR TORQUE SWITCH BYPASS SERVICE IN OPEN CIRCUIT.

DEVICE	MFR	TYPE	MODEL NO.	REFERENCE DWGS NUMBER	REFERENCE DWGS DESCRIPTION	REV
74	C-H	D26M	D26MR13A	1X4DB161-2	P & I DIAGRAM	—
AX	C-H	D26M	D26MR31A	1X3D-AA-F16A	ONE LINE DIAGRAM	—
				1X5DN121-2	LOGIC DIAGRAM	—
				1X5AC01-251	BECHTEL V/P	—

NUCLEAR SAFETY RELATED A

AUXILIARY FW PUMP P4003 DISCHARGE TRAIN A VALVE Q

SOUTHERN COMPANY SERVICES, INC.
BIRMINGHAM, ALABAMA

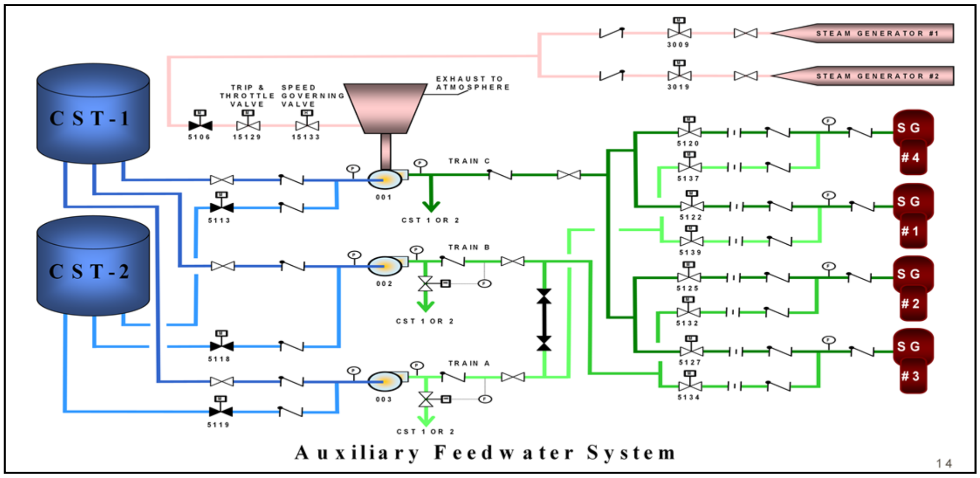
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

ELEMENTARY DIAGRAM
AUXILIARY FEEDWATER SYSTEM
1HV-5137

NO.	REVISIONS	DATE	DR	CHK	APPV	DTL
8	REVISED FOR REA 02-VAA615	06/26/02	JGM	SM/DRD	EOG	WFP

SCALE: NONE	DRAWING NO.	REV.
JOB NO.10604	1X3D-BC-F08C	8

SIZE E 34X44



Objectives 1 & 2

- **Review functions of system – ESF heat sink/feedwater source when MFW not in service.**
- **Review major flowpaths from each CST through pumps into each SG.**
- **Discuss power supplies for each pump and MOV**
  - **Steam supplies for TDAFW pump from A(3009) & B(3019) Train DC power.**
  - **All other TDAFW MOVs (and speed controller) are Train C DC power.**
  - **MDAFW pumps and MOVs are Train related 4160/480VAC power.**
- **Show control room indications and controls**
- **Discuss normal standby alignment of system modes 1, 2, 3.**

AFW is used for startup (up to 4% power) for maintaining SG level.

It is also designed for SG level maintenance following a reactor trip.

(V-LO-PP-20101-01)

After swapping to MFPs in Mode 2 (before transition to Mode 1,) AFW is put in a standby alignment:

-TDAFW may already be aligned but MDAFW cannot be aligned when being used for SGWL control.

-MDAFW pump switches in AUTO.

-TDAFW HV-5106 is shut with the Trip and Throttle valve latched and open.

-Both TDAFW steam supply valves are OPEN.

-TDAFW speed controller demand at maximum.

-All TDAFW and MDAFW discharge valves FULLY OPEN.

(V-LO-PP-20101-02)

1. G2.3.13 001/LOIT/RO/C/A 3.4/3.8/G2.3.13/LO-LP-63308-02///

Initial conditions:

- Unit 1 is in Mode 5.
- RCS level has been lowered to 192 feet to remove reactor head.
- ALARA briefing in progress for closing Equipment Hatch.

Current conditions:

- The Unit Operator is performing a surveillance that specifies an Independent Verification (IV) of a manual valve inside containment.

- The area where the valve is located has the following conditions:
  - Airborne contamination is 0.1 DAC.
  - The operator is expected to receive 15 mrem of dose while performing the IV.

Per NMP-GM-005-002, "Human Performance Tools Instruction," which one of the following identifies the reason for waiving the IV?

A. Plant Mode of operation.

B. RCS at reduced inventory.

C.✓ A significant radiation exposure.

D. Airborne contamination in excess of limits.

K/A

G2.3.13 **Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.**

K/A MATCH ANALYSIS

The question tests the candidate's knowledge of containment entry requirements and radiological safety procedures associated with the Independent Verification process as it applies to excessive dose while performing IVs.

EXPLANATION OF REQUIRED KNOWLEDGE

NMP-GM-005-002 allows for an exemption to the Independent Verification requirement associated with return to service of safety related components based on significant radiological exposure. Step 5.2.2.a.b(2) states IV's may be waived for the following reasons:

- In cases that involve significant radiation exposure.

- In cases that involve containment entry, while containment integrity is established.

NMP-GM-005-002 Definition 3.7 describes Significant Radiation Exposure as greater than or equal to 10mrem whole body dose or airborne contamination in excess of ALARA guidelines. HP ADMIN procedure 00930-C Definition 2.1.b defines areas with an airborne concentration of >0.3 DAC as being in excess of ALARA guidelines for general entries.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. NMP-GM-005-002 makes no provisions for exemption of IVs based on Mode. However, prior to NMP-GM-005-002, the Vogtle specific procedure had an exemption for Modes 1-4 with containment integrity established. A candidate without specific knowledge of the exemption requirements may remember an association with Modes and find it reasonable that IVs would not be required in a lower Mode. In general, as the plant enters Mode 5, many of the safety-related systems are no long required to be maintained OPERABLE. Therefore, this assumption could appear to be justified by Tech Specs.

B. Incorrect. Plausible. NMP-GM-005-002 makes no provisions for exemption of IVs based on RCS inventory. Access to containment and allowed work activities are restricted during these times due to the increased nuclear risk. Additionally, non essential personnel are restricted from access to areas of containment during a head lift due to dose rates. A candidate without specific knowledge of the exemptions found in NMP-GM-005-002 may find it reasonable to exempt the IVs under these circumstances. However, in these cases, the IV would not be exempted it would be delayed and performed later, when conditions were more favorable.

C. Correct. Per NMP-GM-005-002, an IV may be exempted if a dose of greater than or equal to 10 mrem whole body is anticipated to be received.

D. Incorrect. Plausible. NMP-GM-005-002 makes a provision for exemption of IVs based on airborne contamination as a significant exposure. However, airborne contamination would have to be in excess of 0.3 DAC before this would be considered significant. However, a candidate without specific knowledge of IV exemptions may conclude that any airborne contamination would be significant and therefore an exemption would be allowed.

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.3.13	
Importance Rating:	3.4 / 3.8	
Technical Reference:	00930-C, Rev 26.0, page 2 NMP-GM-005-002, Rev 2.0, pages 4, 16, & 17	
References provided:	None	
Learning Objective:	LO-LP-63308-01	Briefly describe the independent verification policy. Include a discussion of the types of verification that are available including concurrent verification.
	LO-LP-63308-02	With regards to independent verification, describe the following: a. conditions which warrant independent verification b. components/systems that require independent verification (and exceptions) c. safety-related, as applicable to independent verification
Question origin:	BANK	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.12	
Comments:		

You have completed the test!

1.0 PURPOSE

The purpose of this instruction is to identify the core set of Human Performance error reduction tools that will be applied at all Southern Nuclear Operating Company (SNC) sites. It identifies when the Human Performance tools will be used, the error precursors that may lead up to the event, the consequences of not using the human performance tools, and at risk practices to avoid.

2.0 APPLICABILITY

This procedure applies to the entire SNC workforce, site and corporate, as well as contractors and vendors who perform work at SNC facilities.

3.0 DEFINITIONS

3.1 **At-risk Practice** – A behavior, belief, assumption, or condition that tends to diminish the effectiveness of the tool.

3.2 **Critical Activity** –An activity, if performed improperly, will cause irreversible harm to plant equipment or people or will significantly impact plant operations.

3.3 **Critical Step** – A procedure step, series of steps, or action that will cause irreversible harm to plant equipment or people or will significantly impact plant operation if performed improperly.

3.4 **Error-Likely Situation** - A work situation in which there is greater opportunity for error when a specified action or task is performed due to the presence of error precursors.

3.5 **Error-Precursors (Risk Factors)** – Task related conditions related to a specific activity or task that provoke human error, increase the chance of a technical error or an adverse consequence.

3.6 **Latent Error** – An error, act, or decision that unknowingly creates an undesired condition(s). It may be embedded in a process, culture, plant configuration of systems, structures, or components or the design bases, or reduces equipment reliability that remains undetected until revealed by subsequent operational activities.

3.7 **Significant Radiation Exposure** - As applicable to activities described by this procedure, greater than or equal to 10 mrem whole body dose or airborne contamination in excess of ALARA guidelines.

0.3 DAC required per 00930-C, RADIATION AND CONTAMINATION CONTROL

4.0 RESPONSIBILITIES

Responsibilities for this instruction listed in NMP-GM-005 RESPONSIBILITIES section.

5.2.1 Pre-Job Brief (continued)

3. (continued)

g. During the Pre-Job Briefing, a determination for a post-job review should be determined and communicated by the job supervisor. The communication will include the time and location of the Post-Job Review as applicable. Note that a Post-Job Review may also be required by NMP-DP-001, "Operational Risk Awareness." See Attachment 4 for items to be covered in a Post-Job Review.

5.2.2 Verification Practices

Verification practices refers broadly to three tools—concurrent verification, independent verification, and peer-checking—that involve a second person to confirm the actions and results achieved by a performer. While peer-checking (PC) focuses on preventing a mistake by the performer, independent verification (IV) and concurrent verification (CV) focus more on confirming the correct configuration, or status, of equipment.

1. Independent Verification

Independent verification is a series of actions by two individuals working independently to confirm the condition of a component after the original act that placed it in that condition.

a. The Purpose of the Independent Verification (IV) tool:

The IV process confirms the condition of equipment required to be in a particular condition to maintain the plant's physical configuration required for safe operation. Otherwise, adverse consequences could result later if the improper condition remains undetected. IV can only be used when an immediate, adverse consequence of a mistake by the performer cannot occur, because IV catches errors after they have been made, not before or during.

NOTE

Independent Verification is performed by an individual who has basic knowledge of the type of component involved (valve, breaker, etc.). The individual need not be trained on the activity or system involved.

b. When to perform Independent Verification:

(1) Independent Verification is required for restoration of:

- Safety related systems or components.
- Valve positions in liquid or gaseous radioactive waste systems that if mis-positioned could lead to unintended or unmonitored radio activity release.

5.2.2 Verification Practices (continued)

1. b. (1) (continued)

- Other component positioning as determined necessary by the Operations Director

(2) Exemption – IV may be waived for the following reasons:

- In cases that involve significant radiation exposure.
- In cases that involve containment entry (PWR) or drywell entry (BWR), while containment integrity is established.

NOTE
Performer and verifier should be dispatched separately.

c. How to perform Independent Verification

(1) The performer of the component manipulation is **SEPARATED** from the verifier by time and distance.

(2) The performer shall, with the use of the controlling document:

(a) **LOCATE** the component and identify each unique identifier on the component label.

(b) **PERFORM** the intended action.

(3) The verifier shall, with use of the controlling document:

(a) **LOCATE** the component and identify each unique identifier on the component label.

(b) **CONFIRM** the completed action.

NOTE
Concurrent Verification is performed by an individual qualified for the activity, systems, and/or components involved and the relationship of these activities, components, and systems to plant safety.

2. **Concurrent Verification**

Concurrent verification (CV) is a series of actions by two individuals working together at the same time and place to separately confirm the condition of a component before, during, and after an action, when the consequences of an incorrect action would lead to immediate and possibly irreversible harm to the plant or personnel.

1.0 PURPOSE(1984301253) (1985303534)

This procedure establishes requirements and responsibilities for monitoring and controlling exposure to radiation and contamination. It includes criteria for Radiation Controlled Areas, the Radiation Work Permit (RWP) system, sampling and surveys, shielding, and the Self Monitoring process as follows:

4.0 REQUIREMENTS
5.0 PROCEDURE
5.1 POSTING
5.2 RADIATION WORK PERMIT
5.3 CONTAMINATION CONTROLS
5.4 PERSONNEL MONITORING
5.6 SURVEYS AND SAMPLING
5.7 SHIELDING
5.8 SELF MONITORING

2.0 DEFINITIONS

2.1 AIRBORNE RADIOACTIVITY AREA (1985306088) (1993327366)

A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations:

a. To such a degree that an individual present in the area without respiratory protection equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake (ALI) or 12 Derived Air Concentrations (DAC)-hours.

OR

b. An area should be considered an Airborne Radioactivity Area when the airborne activity reaches 0.3 DACs for isotopes which have a classification other than submersion, and 100% of the DAC limits for isotopes which have a classification of submersion. DAC limits are specified in 10CFR20 Appendix B, Table I, Column 3.

2.2 ANNUAL LIMIT ON INTAKE (ALI)

The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI values for intake by ingestion and by inhalation of selected radionuclides are given in 10CFR20, Appendix B, Table 1, Columns 1 and 2. (1993327366)

1. G2.3.15 001/LOIT/RO/C/A 2.9/2.9/G2.3.15/LO-LP-60309-08///

Initial condition:

- Unit 1 is at 11% reactor power with a startup in progress.

Current condition:

- ALB15-C01 HIGH RADIATION ALARM is received due to SG tube leakage.

Which one of the following completes the following statement?

1RE-0724, Steam Line Radiation, __(1)__ available to monitor SG activity,

and

the monitor is designed to detect __(2)__ radiation.

	__(1)__	__(2)__
A.	is	noble gas
B.	is	nitrogen-16
C.	is NOT	noble gas
D.✓	is NOT	nitrogen-16

K/A

G2.3.15 **Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.**

K/A MATCH ANALYSIS

The KA addresses the relationship between fixed radiation monitors and their response to alarms. This question is generic in nature because it addresses the operation of a rad monitor that is common to both units.

EXPLANATION OF REQUIRED KNOWLEDGE

Tube Leakage Rad Monitor RE-724 utilizes N-16 to detect and quantify primary to secondary leakage through the SG tubes. N-16 is produced by the fission process at approximately 16% RTP and above. The power level at which N-16 is produced is not exactly 16% RTP, however the detector has proven reliable at greater than or equal to 16% RTP. The rad monitor will indicate at power levels lower than 16%, but will be inaccurate. For this reason, UOP 12004-C step 4.2.28 directs the removal of RE-724 from service at less than 16% power. In contrast, RE-810 and RE-12839 monitor noble gases and are available at lower power levels, although they are not as accurate as

RE-724, . RE-810 monitors SG steam flow and RE-12839 monitors the steam packing and steam jet air ejector flows.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. RE-724 is not accurate and therefore removed from service with reactor power <16% RTP. However, RE-810 is commonly confused with RE-724 and would be available and accurate at 11% RTP.

The second part is incorrect. RE-724 utilizes N-16 for detection. However, other secondary rad monitors utilize noble gas, including RE-0810 and RE-12839. RE-724 and RE-810 are commonly confused by candidates.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. RE-724 utilizes N-16 for detection.

C. Incorrect. Plausible. The first part is correct. RE-724 is not accurate and therefore removed from service with reactor power <16% RTP.

The second part is incorrect. See the second part of choice A above.

D. Correct. The first part is correct. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.3.15	
Importance Rating:	2.9 / 2.9	
Technical Reference:	ARP 17100-1, Rev 26.2, page 42 UOP 12004-C, Rev 107.1, page 56	
References provided:	None	
Learning Objective:	LO-LP-60309-08	Describe how to obtain plant computer readings that provide trend displays that are useful to determine leak rate and rate of leak changes.
	LO-PP-21101-16	Discuss why and where radiation monitors are, install on the Main Steam System.
Question origin:	BANK	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.11 / 43.4 / 45.9	

Comments:

You have completed the test!

INITIALS

> NOTE
>
> Failure of the Main Generator output breakers to open should not be considered an abnormal response requiring initiation of 19000-C, "E-0 Reactor Trip or Safety Injection." ☐

(4) **Trip** the Turbine per 13800, "Main Turbine operation." ________

- **Verify** turbine tripped per 13800, "Main Turbine operation". ________
- **Verify** the Main Generator PCBs are OPEN. per 13800, "Main Turbine operation". ________

4.2.27 At recorder NR-45:

- **Verify** all Power Range channels indicating properly. ________
- **Verify** all Intermediate Range channels indicating properly. ________

4.2.28 **IF** reactor power will be less than 16% for at least a week, **notify** chemistry to **remove** N-16 Rad Monitor RE-724 from service. ________

> NOTE
>
> Step 4.2.29 should only be performed with Shift Manager authorization. ☐

4.2.29 IF authorized by Shift Manager, **place** the Auxiliary Feedwater System in service on mini-flow, per 13610, “Auxiliary Feedwater System.” ________

> NOTE
>
> Shutting down reactor by manually driving All Rods IN should not be performed IF the ARO position was **NOT** verified to be 228 in step 4.2.4. ☐

4.2.30 **IF** the reactor shall be shut down by manually driving ALL RODS IN, **go directly to** step 4.2.34. ________

4.2.31 **IF** the reactor shall be shut down by performing a manual reactor trip, continue with step 4.2.32. ________

ORIGIN	**SETPOINT**	
Steam Line (N16 monitor)	As determined by Chemistry Department	1-RE-0724 (High)

NOTES

- For other than HIGH conditions see Pages 4 and 5. ☐
- This detector monitors secondary activity at power and indicates primary to secondary leakage in GPD and GPD/HR. ☐

1.0 **PROBABLE CAUSE**

Steam Generator Tube leakage

2.0 **AUTOMATIC ACTIONS**

NONE

3.0 **INITIAL OPERATOR ACTIONS**

NONE

1. G2.3.5 001/LOIT/RO/M/F 2.9/2.9/G2.3.5/LO-TA-23005///013K1.13

Initial condition:

- Unit 1 and Unit 2 are at 100% reactor power.

Current conditions:

- Unit 1, ALB05-C03 HIGH RADIATION ALARM is received.
- 1RE-12116, Control Room Air Intake Radiogas Monitor (Train 'A'), RED indication light is illuminated on the SRDC.

- Unit 2 has NO radiation monitor in alarm.

Which one of the following completes the following statement?

The **Unit 1**, Train 'B' CREFS unit __(1)__ automatically start,

and

the **Unit 2**, Train 'B' CREFS unit __(2)__ automatically start.

	__(1)__	__(2)__
A.	will	will
B.✓	will	will NOT
C.	will NOT	will
D.	will NOT	will NOT

K/A

Generic

G2.3.5 **Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.**

K/A MATCH ANALYSIS

The question tests the candidate's ability to use Control Room Intake Rad Monitors 1/2RE-12116/12117 to determine expected operation/response.

EXPLANATION OF REQUIRED KNOWLEDGE

Per ARP 17102-1, 1RE-12116 red lamp indicates a HIGH alarm. The Control Room Ventilation is expected to switch to the Post Accident Mode. Per step 4.1, if the switch did not occur automatically, a manual actuation is to be performed using SOP 13301-C. Per SOP 13301-C section 4.4.1.1, placing either CRI handswitch to ACTUATE will start

the (LEAD) Train B CREFs filter unit on Unit 1 ONLY. Section 4.4.1.2 must be performed to actuate the Unit 2 CREFs system if necessary. Additionally, P&IDs AX4DB206-1 &3 also show that 1/2RE-12116/12117 feed separate Train and separate Unit CRI Logic circuits. Therefore, 1RE-12116 in HIGH alarm will only start the Unit 1 Train B CREFs unit.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. Per ARP 17102-1 and SOP 13301-C section 4.4.1.1, 1RE-12116 in HIGH alarm is expected to result in an automatic start of the Unit 1 Train B CREFs unit.

The second part is incorrect. Per ARP 17102-2, SOP 13301-C section 4.4.1.1 and P&ID AX4DB206-1 &3, 1RE-12116 in HIGH alarm will NOT result in an automatic start of the Unit 2 Train B CREFs unit. Actuation of the Unit 2 B Train CREFs unit would come from either 2RE-12116 or 2RE-12117 in HIGH alarm. However, since the Unit 1 and Unit 2 Control Rooms share a common envelope, it is reasonable to believe that an actuation of one Unit's intake rad monitor would actuate CRI on both Units.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. Per ARP 17102-2, SOP 13301-C section 4.4.1.1 and P&ID AX4DB206-1 &3, 1RE-12116 in HIGH alarm is NOT expected to result in an automatic start of the Unit 2 Train B CREFs unit. Actuation of the Unit 2 B Train CREFs unit would come from either 2RE-12116 or 2RE-12117 in HIGH alarm.

C. Incorrect. Plausible. The first part is incorrect. Per ARP 17102-1 and SOP 13301-C section 4.4.1.1, 1RE-12116 in HIGH alarm is expected to result in an automatic start of the Unit 1 Train B CREFs unit. However, a candidate could find it reasonable that 1RE-12116 would actuate Train A CREFs and not Train B CREFs.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is correct. See the second part of choice B above.

You have completed the test!

Level:	RO	
Tier # / Group #	T3	
K/A#	G2.3.5	
Importance Rating:	2.9 / 2.9	
Technical Reference:	P&ID AX4DB206-1, Rev 31.0 P&ID AX4DB206-3, Rev 29.0 SOP 13301-C, Rev 30.0, pages 27-34 ARP 17102-1, Rev 20.2, pages 23 & 24	
References provided:	None	
Learning Objective:	LO-TA-23005	Manual actuation of Control Room Isolation using 13301-C
	LO-TA-32007	Verify Proper Automatic actions to high radiation alarms
	LO-PP-23301-03	Describe the Emergency and Isolation mode of operation for the Control Room HVAC system including flow paths and interlocks.
	LO-PP-23301-05	List the actuating signals for CRI including indications and automatic actions.
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.7 / 41.11 / 41.12	

Comments: Question appears to match the KA, but is not plant-specific and is not at the RO level. Anyone qualified to work in the RCA should know the answer to this question.

I recommend replacing this question. Given that the KA is generic and broad (ability to use radiation monitoring systems pretty much lends itself to any detector used to measure radiation), I would ask a question where a radiation monitor is alarming (during fuel handling, during operations, while shut down, etc) and ask the required actions (can be out of an AOP or annunciator response) or ask if a particular AOP is required to be entered (entry conditions for AOPs and EOPs is RO knowledge). This question can be similar to any questions that may also fit a process radiation monitor question. The key is to keep it plant specific and at the level of an operator.

-JAT 12/19/2013

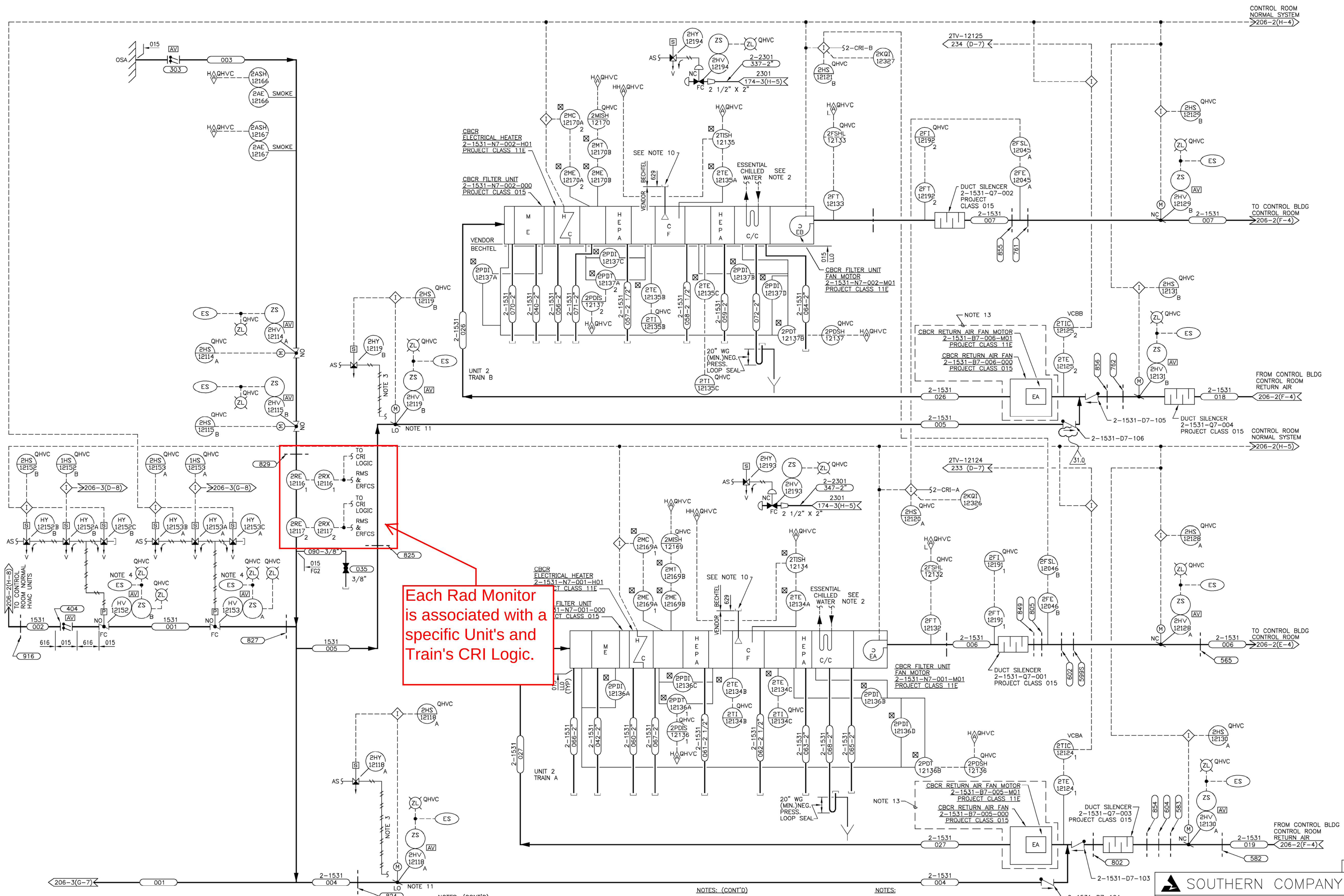
Question is plant specific and appears to match the KA. One thing to note is that there are four distinct answers just with the first part of the question – the part regarding the Essential chillers does not appear needed to answer the question. Additionally, since choice A is not

specific as to which train of CREFS starts, it is an outlier. Really, what you're asking is which train of CREFS starts and on which units. That can be done by asking:

Unit 1 Train "A" CREFS [does/does not] start and Unit 2 Train "A" CREFS [does/does not] start.

Or something similar… i.e., make it a 2x2, soliciting the answer to two pieces of information, since that's what the question ultimately boils down to. It doesn't have to be asked like the example I gave; that's just a starting point. This way, question psychometrics can't be used to whittle the choices down, and you also don't use the word ONLY, which can be problematic.
-JAT 2/4/14

You have completed the test!



NOTES: (CONT'D)

11. PROVIDE FOR POWER LOCKOUT AND RESTORATION FROM CONTROL ROOM.
12. DELETED
13. THE RETURN AIR FANS 2-1531-B7-005 & 2-1531-B7-006 ARE RETIRED IN PLACE WITH THE BLADES REMOVED.

NOTES: (CONT'D)

5. REFER TO FLOW DIAGRAM AX4DB255-1.
6. DELETED
7. DELETED
8. DELETED
9. DELETED
10. FIRE HOSE CONNECTION (FEMAILE NPT X 1 1/2" FEMALE NHT COUPLING WITH 1 1/2" NHT PLUG.)

NOTES:

1. DELETED
2. FOR ESSENTIAL CHILLED WATER TRAIN "A" SEE DWG 2X4DB233. FOR TRAIN "B" SEE DWG 2X4DB234.
3. CONTROL AIR TO INFLATABLE SEAL ON BUBBLETIGHT DAMPER.
4. POSITION INDICATION OF THESE COMMON VALVES IS INPUT TO ONLY THE UNIT 1 COMPUTER.

CAD AX4DB206-1 | ACAD2002 | BgT - 02

NO.		DATE	DR	CHK	APPV	DTL
31.0	REVISED PER ABN-V00745, VER.1.0	06-13-07	BgT	LRP	WLN	

SOUTHERN COMPANY

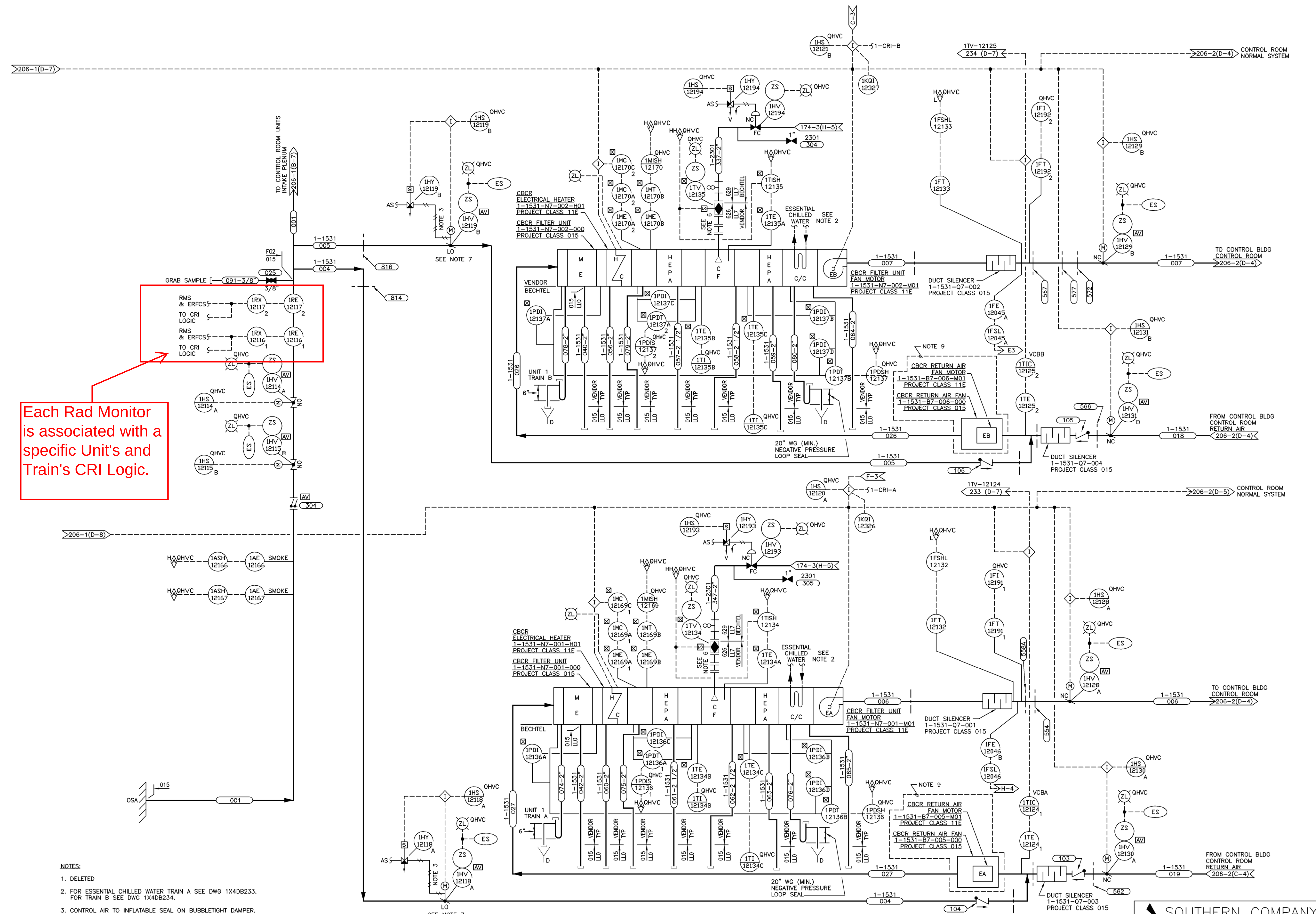
GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CONTROL BUILDING CONTROL ROOM
HVAC SYSTEM LEVEL 1
SYSTEM NO. 1531

SCALE: NONE | JOB NO.10604 | DRAWING NO. AX4DB206-1 | VER. 31.0

SIZE E 34X44

DRAWING CATEGORY: CRITICAL



NOTES:

1. DELETED
2. FOR ESSENTIAL CHILLED WATER TRAIN A SEE DWG 1X4DB233. FOR TRAIN B SEE DWG 1X4DB234.
3. CONTROL AIR TO INFLATABLE SEAL ON BUBBLETIGHT DAMPER.
4. DELETED
5. REFER TO FLOW DIAGRAM AX4DB255-1.
6. DELUGE VALVES ARE DISABLED SO THAT VALVE OPENS WITH WATER PRESSURE. REMOVABLE SPOOL PIECE IS REMOVED FOR NORMAL OPERATION AND INSTALLED TO DELUGE CARBON FILTER.
7. PROVIDE FOR POWER LOCKOUT AND RESTORATION FROM CONTROL ROOM.
8. DELETED
9. RETURN AIR FANS 1-1531-B7-005 AND 1-1531-B7-006 ARE RETIRED IN PLACE WITH THE BLADES REMOVED.

NO.	VERSIONS	DATE	DR	CHK	APPV	DTL
29.0	REDRAWN PER ABN-V70113, VER. 1.0	08/21/06	NJ	LCF	EPD	X

CAD AX4DB206-3
ACAD2002 NJ - 01

SOUTHERN COMPANY

GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT

P & I DIAGRAM
CONTROL BUILDING CONTROL ROOM
HVAC SYSTEM LEVEL 1
SYSTEM NO. 1531

SCALE: NONE
JOB NO.10604
DRAWING NO. AX4DB206-3
VER. 29.0
SIZE E 34X44
DRAWING CATEGORY: CRITICAL

INITIALS

4.4 NON-PERIODIC OPERATIONS

4.4.1 Manual Actuation Of Control Room Isolation

NOTES

- This section is written using Unit 1, Unit 2 and Common component designations. Some Unit 2 designations are shown in parenthesis. ☐
- If the TRAIN B CR FLTR UNIT SUPPLY FAN fails to start on actuation, the Train A Fan will start after a 30 second time delay. ☐
- The TSC Air Filtration System will automatically start on manual initiation of Control Room Isolation. ☐

ALB05-D05 GROUP 4 MONITOR LIGHT COMP OFF NORM	☐
ALB39-D05 480V SWGR ANB30 TROUBLE	☐
ALB50-B03 CR HI/LO DIFF PRESS	☐

4.4.1.1 Manually initiate Control Room Isolation on Unit One:

a. **Place** either CR ISO MANUAL ACTUATION Switch in ACTUATE:

- 1-HS-12195A [A4] (TRAIN A) to ACTUATE. ________
- 1-HS-12196A [A6] (TRAIN B) to ACTUATE. ________

b. **Verify** that TRAIN B CR FLTR UNIT SUPPLY FAN (LEAD), 1-1531-N7-002 [B10] starts. ________

c. **Verify** that TRAIN A CR FLTR UNIT SUPPLY FAN, 1-1531-N7-001 [B8] (STANDBY) does NOT start: ________

Placing either CRI actuation handswitch on Unit 1 to ACTUATE starts the B Train CREFs units on Unit 1 only. Unit 2 must be actuated separately by section 4.4.1.2.

INITIALS

d. **Verify** that both KIT TOIL + CONF RM EXH ISO DMPRs close:

A-HV-12162 [D6], TRAIN A, **CLOSED** _________

A-HV-12163 [D7], TRAIN B, **CLOSED** _________

NOTES

- All positions in Step 4.4.1.1.e may be verified in any order. ☐
- Unit Two dampers are in bold print to help in identification. ☐

e. **Verify** the following damper positions;

(1) CR NORM AIR SUPPLY ISO DMPRs,

1-HV-12146 [C6], TRAIN A CLOSED _________

2-HV-12146 [C6], TRAIN A CLOSED _________

1-HV-12147 [C7], TRAIN B CLOSED _________

2-HV-12147 [C7],, TRAIN B CLOSED _________

(2) CR NORM AIR RTN ISO DMPRs,

1-HV-12149 [E6] TRAIN A CLOSED _________

2-HV-12149 [E6] TRAIN A CLOSED _________

1-HV-12148 [E7] TRAIN B CLOSED _________

2-HV-12148 [E7] TRAIN B CLOSED _________

INITIALS

f. **Verify** the following for the Filter Unit that started;

(1) IF Train B Filter Unit, 1-1531-N7-002 [B10] started,

(a) **Verify** that the Train B CR FILTER UNIT OUTLET AIR DMPR, 1-HV-12129 [C11] OPEN. ________

(b) **Verify** that the Train B CR RTN FAN INLET AIR DMPR on the running train, 1-HV-12131 [D10] OPEN. ________

(c) **Verify** that the Train B CR NORMAL HVAC UNIT INTAKE ISO DMPR on the running train, A-HV-12152 [B7] CLOSED. ________

(2) IF Train A Filter Unit, 1-1531-N7-001 [B8] started,

(a) **Verify** that the Train A CR FILTER UNIT OUTLET AIR DMPR, 1-HV-12128 [C9] OPEN. ________

(b) **Verify** that the Train A CR RTN FAN INLET AIR DMPR on the running train, 1-HV-12130 [D8] OPEN. ________

(c) **Verify** that the Train A CR NORMAL HVAC UNIT INTAKE ISO DMPR on the running train, A-HV-12153 [B6] CLOSED. ________

g. **Verify** that the CR NORM AC UNIT SUPPLY FANs, shut down.

- A-1531-A7-001 [C4], STOPPED. ________
- A-1531-A7-002 [C5], STOPPED. ________

h. **Verify** that the CR NORM AC UNIT EXH FANs, shut down.

- A-1531-B7-009 [D4] STOPPED. ________
- A-1531-B7-010 [D5], STOPPED. ________

i. **Verify** that the KITCH TOILET AND CONF RM EXH FAN, A-HS-12164 in the Shift AA's Office, stops. ________

INITIALS

NOTE

If it is necessary to isolate outside air to the Control Room in the next step due to smoke or toxic gas intake, both the Unit 1 and Unit 2 dampers should be shut. ☐

j. IF Control Room outside air is restricted for Control Room habitability due to smoke or toxic gas intake, THEN **close** the CR Outside Air Supply Dampers for BOTH Units:

UNIT 1

1-HS-12114 [E8] ________

1-HS-12115 [E10] ________

UNIT 2

2-HS-12114 [E8] ________

2-HS-12115 [E10] ________

k **Verify** proper operation of the TSC Air Filtration System per 13303-C, "Technical Support Center And Central Alarm Station HVAC Systems." ________

l. **Verify** proper Essential Chiller operation. ________

INITIALS

NOTES

- This section is written using Unit 1, Unit 2 and Common component designations. Some Unit 2 designations are shown in parenthesis. ☐
- If the TRAIN B CR FLTR UNIT SUPPLY FAN fails to start on actuation, the Train A Fan will start after a 30 second time delay. ☐
- The TSC Air Filtration System will automatically start on manual initiation of Control Room Isolation. ☐

ALB05-D05 GROUP 4 MONITOR LIGHT COMP OFF NORM ☐

ALB39-D05 480V SWGR ANB30 TROUBLE ☐

ALB50-B03 CR HI/LO DIFF PRESS ☐

4.4.1.2 Manually initiate Control Room Isolation on Unit Two:

a. **Place** either CR ISO MANUAL ACTUATION Switch in ACTUATE:

- 2-HS-12195A [A4] (TRAIN A) to ACTUATE. ________
- 2-HS-12196A [A6] (TRAIN B) to ACTUATE. ________

b. **Verify** that TRAIN B CR FLTR UNIT SUPPLY FAN (LEAD), 2-1531-N7-002 [B10] starts. ________

c. **Verify** that TRAIN A CR FLTR UNIT SUPPLY FAN, 2-1531-N7-001 [B8] (STANDBY) does NOT start: ________

d. **Verify** that both KIT TOIL + CONF RM EXH ISO DMPRs close:

A-HV-12162 [D6], TRAIN A, **CLOSED** ________

A-HV-12163 [D7], TRAIN B, **CLOSED** ________

INITIALS

NOTES

- All positions in Step 4.4.1.2.e may be verified in any order. ☐
- Unit Two dampers are in bold print to help in identification. ☐

e. **Verify** the following damper positions;

(1) CR NORM AIR SUPPLY ISO DMPRs,

1-HV-12146 [C6], TRAIN A CLOSED ________

2-HV-12146 [C6], TRAIN A CLOSED ________

1-HV-12147 [C7], TRAIN B CLOSED ________

2-HV-12147 [C7],, TRAIN B CLOSED ________

(2) CR NORM AIR RTN ISO DMPRs,

1-HV-12149 [E6] TRAIN A CLOSED ________

2-HV-12149 [E6] TRAIN A CLOSED ________

1-HV-12148 [E7] TRAIN B CLOSED ________

2-HV-12148 [E7] TRAIN B CLOSED ________

f. **Verify** the following for the Filter Unit that started;

(1) IF Train B Filter Unit, 2-1531-N7-002 [B10] started,

(a) **Verify** that the Train B CR FILTER UNIT OUTLET AIR DMPR, 2-HV-12129 [C11] OPEN. ________

(b) **Verify** that the Train B CR RTN FAN INLET AIR DMPR on the running train, 2-HV-12131 [D10] OPEN. ________

(c) **Verify** that the Train B CR NORMAL HVAC UNIT INTAKE ISO DMPR on the running train, A-HV-12152 [B7] CLOSED. ________

INITIALS

(2) IF Train A Filter Unit, 2-1531-N7-001 [B8] started,

(a) **Verify** that the Train A CR FILTER UNIT OUTLET AIR DMPR, 2-HV-12128 [C9] OPEN. ________

(b) **Verify** that the Train A CR RTN FAN INLET AIR DMPR on the running train, 2-HV-12130 [D8] OPEN. ________

(c) **Verify** that the Train A CR NORMAL HVAC UNIT INTAKE ISO DMPR on the running train, A-HV-12153 [B6] CLOSED. ________

g. **Verify** that the CR NORM AC UNIT SUPPLY FANs, shut down.

- A-1531-A7-001 [C4], STOPPED. ________
- A-1531-A7-002 [C5], STOPPED. ________

h. **Verify** that the CR NORM AC UNIT EXH FANs, shut down.

- A-1531-B7-009 [D4] STOPPED. ________
- A-1531-B7-010 [D5], STOPPED. ________

i. **Verify** that the KITCH TOILET AND CONF RM EXH FAN, A-HS-12164 in the Shift AA's Office, stops. ________

INITIALS

> NOTE
>
> If it is necessary to isolate outside air to the Control Room in the next step due to smoke or toxic gas intake, both the Unit 1 and Unit 2 dampers should be shut. ☐

j. IF Control Room outside air is restricted for Control Room habitability due to smoke or toxic gas intake, THEN **close** the CR Outside Air Supply Dampers for BOTH Units::

UNIT 1

1-HS-12114 [E8] ________

1-HS-12115 [E10] ________

UNIT 2

2-HS-12114 [E8] ________

2-HS-12115 [E10] ________

k **Verify** proper operation of the TSC Air Filtration System per 13303-C, "Technical Support Center And Central Alarm Station HVAC Systems." ________

l. **Verify** proper Essential Chiller operation. ________

WINDOW CDCA B5

ORIGIN	**SETPOINT**	
Control Room Air Intake Process Radiogas Monitor 1-RE-12116	As determined by Chemistry Department	1-RE-12116 (RED LAMP LIT) (HIGH)

NOTE

For other than HIGH conditions see Pages 5 and 6. ☐

1.0 PROBABLE CAUSE

1. Gaseous radioactivity in the incoming air.
2. Equipment malfunction.

2.0 AUTOMATIC ACTIONS

Control Room and Technical Support Center Ventilation switches to the Post Accident Mode.

3.0 INITIAL OPERATOR ACTIONS

NONE

4.0 SUBSEQUENT OPERATOR ACTIONS

1. **Verify** a Control Room Filtration Unit is running. If not, manually **start** per 13301-C “CBCR Normal HVAC and Emergency Filtration System”, Section for manually initiating a Control Room Isolation.

2. **Verify** the Chilled Water System associated with the operating Emergency Control Room HVAC train has started per 13744A-1 “Train A Essential Chilled Water System” or 13744B-1 “Train B Essential Chilled Water System” as appropriate.

3. **Verify** Technical Support Center Air Filtration Unit in service per 13303-C “Technical Support Center And Central Alarm Station HVAC Systems”.

4. Refer to 91001-C, "Emergency Classification And Implementing Instructions."

5. **Notify** Health Physics to locate and determine the cause of the contamination.

6. IF sampling and analysis determine that the channel has malfunctioned:

   a. **Comply** with Technical Specifications LCO 3.3.7.

   b. **Place** 1-HS-12195C on QESF to TEST BLOCK CHAN I.

   c. **Request** Chemistry to investigate and take corrective action.

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: AX4DB206-3, 1X3D-BG-C03C, 1X5DX2101

1. G2.4.25 001/LOIT/RO/M/F 3.3/3.7/G2.4.25/LO-PP-43101-02///

Given the following:

- Both Unit 1 and Unit 2 are at 100% reactor power.
- A fire occurs in the Fuel Handling Building as a result of a **seismic** event.

Which one of the following completes the following statement?

The __(1)__ pumps and piping are designed to provide firefighting water following the seismic event,

and

per 13903-C, "Fire Protection System Operation," the flow path __(2)__ automatically initiate water flow without operator action.

	__(1)__	__(2)__
A.	NSCW	will
B.✓	NSCW	will NOT
C.	Diesel fire	will
D.	Diesel fire	will NOT

K/A

G2.4.25 Knowledge of fire protection procedures.

K/A MATCH ANALYSIS

The question addresses the operator's knowledge of fire procedures related to a specific event, the fire protection system that is available, and the action required to place it in service.

EXPLANATION OF REQUIRED KNOWLEDGE

Per the NOTE at the begining of procedure SOP 13150A-1, SOP 13903-C is utilized to place the Seismic Standpipe Fire Water Protection System in service. SOP 13903-C section 4.4 gives direction to unlock and open the manual valves in Table 1 that supply water to the various building seismic standpipes. Step 4.4.1 requires the candidate to determine the inservice NSCW train and use it to supply it's respective standpipe system.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is correct. The pumps and piping available for fire

protection following a seismic event are in the NSCW system.

The second part is incorrect. SOP 13903-C section 4.4 gives direction to unlock and open the manual valves in Table 1 that supply water to the various building seismic standpipes. However, the candidate may believe seismic standpipe system piping is integral to the normal fire water piping and is available without local action in the event of a fire, and will automatically actuate when a fire is present.

B. Correct. The first part is correct. See the first part of choice A above.

The second part is correct. SOP 13903-C section 4.4 gives direction to unlock and open the manual valves in Table 1 that supply water to the various building seismic standpipes.

C. Incorrect. Plausible. The first part is incorrect. The pumps and piping available for fire protection following a seismic event are in the NSCW system. However, the candidate may determine the fire system is designed to class 1 seismic standards since it is protecting buildings with safety related components. In addition, the fire protection system is designed to handle almost all other events, including a loss of offsite power via the diesel powered fire pumps. This could lead the candidate to believe it is available following a seismic event as well.

The second part is incorrect. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the second part of choice C above.

The second part is correct. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T / G	
K/A#	G2.4.25	
Importance Rating:	3.3 / 3.7	
Technical Reference:	SOP 13150A-1, Rev 8.2, page 4 SOP 13903-C, Rev 44.0, pages 38 & 78 thru 80	
References provided:	None	
Learning Objective:	LO-PP-06101-13	Describe how the NSCW system may be used to provide fire protection water following a seismic event at VEGP.
	LO-PP-43101-02	Describe how the Seismic I standpipe is placed in service and the areas protected.
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.4 / 41.8 / 41.10 / 43.5 /45.13	
Comments:		

You have completed the test!

1.0 PURPOSE

NOTE

13903-C, "Fire Protection System Operation" provides direction to place the Seismic Category 1 Standpipe Fire Water Protection System in service.

This procedure provides instructions for operation of the Nuclear Service Cooling Water (NSCW) System including operation of the NSCW Pumps, Cooling Towers, and Cooling Tower Makeup System. Operating instructions are provided in the following subsections:

4.1.1 Train A NSCW Startup to Standby

4.1.2 Train A NSCW Startup from Standby

4.1.3 Train A NSCW Cooling Tower Makeup System Startup

4.2.1 Shifting Train A NSCW Pumps

4.2.2 Shifting NSCW Cooling Tower Makeup Pumps

4.3.1 Train A NSCW Shutdown to Standby

4.4.1 Deicing Train A NSCW Cooling Tower when Operating

4.4.2 Idle NSCW Train Cooling Tower Deicing

4.4.3 Manual NSCW Cooling Tower Makeup to Train A

4.4.4 Alternate NSCW Cooling Tower Makeup to Train A

4.4.5 Makeup to NSCW Cooling Tower A from NSCW Cooling Tower B

4.4.6 Fill and Vent Train A using NSCW Cooling Tower Cross Pumping Provision

4.4.7 Fill and Vent of Train A Safety Related Pump Motor Coolers

4.4.8 Train A NSCW Single Pump Operation (Outage)

4.4.9 Train A NSCW Single Pump Operation (Abnormal)

INITIALS

4.3.4 WHEN directed, **stop** Jockey Pump C-2301-P4-004 as follows:

a. **Check** Jockey Pump C-2301-P4-001(in Electric Pump House) in operation. ________

b. **Place** handswitch C-HS-7902 to OFF on PFH2. ________

4.4 CATEGORY I STANDPIPE OPERATION

NOTES

- The Standpipe System should be used only if the normal Fire Protection System is incapable of fire water delivery. CO1985306084
- Standpipe connections back pack in the Primary Dress Out Locker (CB R104) has two 100 foot sections of 1 1/2” hose, two 1 1/2” nozzles, AND one gated “Y”.
- For additional equipment in the Auxiliary Building, two 2-1/2” fire hoses, three adjustable spray nozzles, bolt cutters, zip ties, etc. can be found in a “JOBOX” located against the south wall of the BTRS chiller room (AB Rm 124)

CAUTION

The design basis for the Standpipe System is two fire hoses operating for 30 minutes at 100 gal/min each. (6000 gal extracted from the Ultimate Heat Sink)

4.4.1 **Determine** which NSCW Train is in service to supply its respective Standpipe System. ________

4.4.2 **Unlock** and **open** the in service NSCW train Standpipe System Supply Valve to the affected building per TABLE 1. ________

4.4.3 **Open** the Fire Hose Valve to the affected area per TABLE 2.

4.4.4 **Notify** the Control Room to monitor NSCW basin level. ________

4.4.5 **Close** all valves opened in Steps 4.4.2 and 4.4.3 WHEN officially secured from the fire event; (IV REQUIRED). ________

TABLE 1

COMPONENT	DESCRIPTION	LOCATION
AUX BLDG		
TRAIN A		
1-1202-U4-089	NSCW AUX BLDG LEVEL B ISO FROM NSCW	R-A40A
2-1202-U4-089	NSCW TRAIN A SPLY TO AB FIRE HOSE CONN	R-B92
TRAIN B		
A-1202-U4-153	NSCW NUC SERV CLG WTR TO AUX BLDG ISO VLV	R-B08(FHB)
1-1202-U4-091	NSCW AUX BLDG LEVEL B ISO FROM NSCW	R-B28
2-1202-U4-091	NSCW TRAIN B SPLY TO AB FIRE HOSE CONN	R-B92
CONTROL BLDG		
TRAIN A		
1-1202-U4-004	NSCW AUX BLDG LEVEL B ISO FROM NSCW (IN OVERHEAD AT FREIGHT ELEVATOR)	1CB316
2-1202-U4-004	NSCW TRAIN A SPLY TO CB FIRE HOSE CONN	2CB328
TRAIN B		
1-1202-U4-003	NSCW SYS INTERTIE TO CB LEVEL 2 FIREWATER (IN OVERHEAD AT FREIGHT ELEVATOR)	1CB316
2-1202-U4-003	NSCW TRAIN B SPLY TO CB FIRE HOSE CONN	2CB308

Sheet 2 of 3

TABLE 1

COMPONENT	DESCRIPTION	LOCATION
CONTAINMENT BLDG		
TRAIN A		
1-1202-U4-002	NSCW TO FIRE PROT HOSE ADAPTOR ISOLATION	EL 207' COL 11
2-1202-U4-002	NSCW TRAIN A SPLY TO CNMT FIRE HOSE CONN	EL 210' COL 11
TRAIN B		
1-1202-U4-001	NSCW TO FIRE PROT HOSE ADAPTOR ISOLATION (7 FT RIGHT OF COLUMN 18)	LEVEL 1
2-1202-U4-001	NSCW TRAIN B SPLY TO CNMT FIRE HOSE CONN	EL 220' COL 19

Sheet 3 of 3

TABLE 1

COMPONENT	DESCRIPTION	LOCATION
DIESEL GENERATOR BLDG		
TRAIN A		
1-1202-U4-028	NSCW SYS HOSE ISO TRAIN A DG BLDG	DG A R-103
2-1202-U4-028	NSCW TRAIN A SPLY TO DG A FIRE HOSE CONN	DG A R-101
TRAIN B		
1-1202-U4-029	NSCW SYS HOSE ISO TRAIN A DG BLDG	DG B R-101
2-1202-U4-029	NSCW TRAIN B SPLY TO DG B FIRE HOSE CONN	DG B R-103

1. G2.4.45 001/LOIT/RO/C/A 4.1/4.3/G2.4.45/LO-TA-60019///

Initial conditions:

- Unit 1 is at 100% reactor power.
- ALB13-A01 STM GEN 1 FLOW MISMATCH is received.

Current conditions:

- RCS pressure is 2237 psig and rising.
- Main generator output is 1225 MWe and rising.

Which one of the following completes the following statement?

A __(1)__ line break is in progress,

and

per 18008-C, "Secondary Coolant Leakage," a main steam line isolation __(2)__ required to be manually actuated following the manual reactor trip.

	__(1)__	__(2)__
A.✓	feed	is
B.	feed	is NOT
C.	steam	is
D.	steam	is NOT

K/A

G2.4.45 **Ability to prioritize and interpret the significance of each annunciator or alarm.**

K/A MATCH ANALYSIS

The questions tests the candidate's ability to interpret annunciator ALB13-A01 STM GEN 1 FLOW MISMATCH utilizing current plant condtions, and respond appropriately.

EXPLANATION OF REQUIRED KNOWLEDGE

ALB13-A01 STM GEN 1 FLOW MISMATCH simply detects a mismatch between steam and feedwater flows in each loop. Plant conditions must be referenced and interpreted to distinguish between the various accidents and malfunctions. The stem gives RCS pressure slightly above normal and rising with Turbine load at 1211 MWe, which is slightly above normal and rising. Both of these conditions are symptoms of a heat up of the RCS, which would either be a decrease in feed water flow or a decrease in steam flow. Since a decrease in steam flow would also cause a reduction in Turbine

load, the event must be a reduction in feed flow by either a break or restriction.

Per 18008-C step 1, if leakage is hazardous to personnel or equipment, the reactor is tripped and a Main Steam Line isolation performed after verification of the reator trip. Then EOP 19000-C, "Reactor Trip or Safety Injection" is entered. The determination of "hazardous" is based on the potential to cause damage or injury. Operators are trained that if a steam or feed leak can be detected using the control room instruments and there is no evidence that the leak is being directed through a tailpipe, then the leak is considered hazardous. No tailpipe flow paths exist for feedwater, therefore the crews' ability to see the response makes it hazardous by definition. This originated from industry OE at Oconee Nuclear Plant, where 8 personnel were killed due to an extraction steam leak in the Turbine Building with the leak not even noticeable in the Main Control Room.

Main steam lines are required to be isolated on a feedline break to prevent blowing down all steam generators once the faulted stream generator is blown down through the connecting steam line piping.

ANSWER / DISTRACTOR ANALYSIS

A. Correct

The first part is correct. ALB13-A01 STM GEN 1 FLOW MISMATCH in conjunction with elevated and rising pressurizer pressure and turbine load are symptoms of a feed line break.

The second part is correct. 18008-C step 1 requires tripping the reactor, verifying the reactor tripped, and isolating steam lines even for feedwater line breaks.

B. Incorrect. Plausible.

The first part is correct. See the first part of choice A above.

The second part is incorrect. 118008-C step 1 requires tripping the reactor, verifying the reactor tripped, and isolating steam lines even for feedwater line breaks. However, it is reaonsable for a candidate to not consider the back flow from the other SGs following blow down and conclude that a MSLI is not required. No steam flow will be seen from the other SGs until the faulted SG blows down all its inventory and creates a pathway for reverse flow. Also, the candidate may believe that the resulting P-4/Lo Tavg automatic feedwater isolation resulting from the reactor trip will isolate the feed line break and a MSLI would not be required.

C. Incorrect. Plausible.

The first part is incorrect. ALB13-A01 STM GEN 1 FLOW MISMATCH in conjunction with elevated and rising pressurizer pressure and turbine load are symptoms of a feed line break. However, a candidate with insufficient diagnostic skills could conclude that an increase in steam flow will result in an increase in reactor power and erroneously conclude that the increase in reactor power will result in an increase in RCS temperature. Additionally, 18008-C is most commonly entered for steam line leaks during scenario training.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

LLevel:	RO
Tier # / Group #	T3
K/A#	G2.4.45
Importance Rating:	4.1 / 4.3

Technical Reference: AOP 18008-C, Rev 9.2, pages 1 & 3
ARP 17001-1, Rev 31.0, page 7

References provided: None

Learning Objective: LO-LP-37121-05 Describe the plant response to the following conditions:

a. steam line break vs feed line break
b. break at end of life vs break at beginning of life core
c. steam break at full power initially vs zero power
d. feed break inside last check valve vs feed break outside last check valve
e. steam break between SG and first MSIV vs steam break between SG and outside last MSIV

LO-TA-60019 Respond to a Loss of Secondary Coolant per 18008-C

Question origin: MODIFIED - HL17 # 040AA1.20

Cognitive Level: C/A

10 CFR Part 55 Content: 41.4 / 41.5 / 41.10 / 43.5 /45.3 / 45.12

Comments:

You have completed the test!

1. 040AA1.20 001/1/1/SLB-CNMT PRESS/TEMP/H-4.1/4.2/NEW/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

Original Question

- Unit 1 is at 100% power.

Current conditions:

- Reactor power is 100.4% and slowly rising.
- RCS pressure is 2212 psig and slowly lowering.
- Turbine load is 1200 MWe and lowering.
- Containment pressure is 1.4 psig and rising.
- Containment temperature is 117.5^{o}F and rising.

Which one of the following correctly completes the following statement?

A ___(1)___ break is the event in progress

and

per 18008-C, "Secondary Coolant Leakage", the FIRST action the operators will perform is to ___(2)___ .

	__(1)__	__(2)__
A.✔	steamline	reduce turbine load
B.	steamline	manually insert rods
C.	feedline	reduce turbine load
D.	feedline	manually insert rods

ABNORMAL OPERATING PROCEDURE
CONTINUOUS USE

PURPOSE

This procedure specifies operator actions for secondary leaks which do NOT actuate Engineered Safeguards Features.

SYMPTOMS

- ALB13-A01 (B01, C01, D01) STM GEN 1 (2, 3, 4) FLOW MISMATCH
- ALB13-A06 (B06, C06, D06) STM GEN 1 (2, 3, 4) HI/LO LVL DEVIATION
- High containment pressure, temperature, moisture, and sump levels WITHOUT radiation.
- High condenser hotwell makeup rates.
- High CST makeup rates.
- Observed secondary leakage.
- Unexplained rise in reactor power.
- Reactor power significantly higher than turbine power.

MAJOR ACTIONS

♦ Evaluate and stabilize plant conditions.

♦ Locate and isolate leakage.

♦ Determine whether to continue operation or initiate plant shutdown.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. Perform the following as necessary:

- Reduce Turbine load if any of the following indications exceed 100% power:

  UQ1118 (GREATER THAN 100% MWT)

  NIs

  ΔTs

- Isolate the leak.

- IF leakage is such that significant hazard to personnel or equipment exists OR leakage rate is unstable and is worsening, THEN:

  1) Trip the reactor.

  2) WHEN reactor trip is verified, THEN close MSIVs and BSIVs.

  3) Go to 19000 - C, E - 0 REACTOR TRIP OR SAFETY INJECTION.

2. Initiate the Continuous Actions Page.

WINDOW A01

DIGITAL FEEDWATER POINT NAME	SETPOINT
DFW1FLOWMM	N/A

STM GEN 1
FLOW MISMATCH

1.0 PROBABLE CAUSE

1. Steam Generator 1 Feedwater Level Control System malfunction.
2. Main Feedwater System malfunction.
3. Feedwater/Steam leak.
4. Main Steam Isolation Valve malfunction.

2.0 AUTOMATIC ACTIONS

IF turbine power is greater than C-20 (40% on PT505 and PT506), an AMSAC actuation will occur after a time delay IF the feedwater flow on any of three loops is less than 25% of normal 100% feedwater flow. The time delay ramps from 27.5 seconds at 100% turbine power to 230 seconds at 40% turbine power.

3.0 INITIAL OPERATOR ACTIONS

1. **Verify** all MSIVs are open. IF any MSIV has failed shut, THEN **trip** the reactor AND **Go To** 19000-C, “E-0 Reactor Trip or Safety Injection”.

2. **Check** Steam Generator 1 feedflow AND steamflow indications:

   a. IF leakage is indicated, **Go To** 18008-C, "Secondary Coolant Leakage".

   b. IF a Main Feedwater System malfunction is indicated, **Go To** 18016-C, "Condensate And Feedwater Malfunction".

3. **Verify** proper operation of 1-FV-4486. IF malfunction is indicated, THEN **take manual control** of 1-FIC-4486 AND **control** as necessary to **maintain** MFPT suction pressure greater than 300 psig AND flow as indicated on 1-FIC-4486 greater than 7400 gpm.

1. WE02EA1.03 001/LOIT AND LOCT/RO/C/A 3.8/4.0/WE02EA1.03/LO-TA-37009///

Initial condition:

- 19010-C, "Loss of Reactor and Secondary Coolant," is in use.

Current conditions:

- Step 11, "Check if ECCS flow should be reduced," is in progress with the following conditions:
  - RCS pressure is 1335 psig and stable.
  - Pressurizer level is 30% and stable.
  - CETCs are 550°F.
  - WR THot is 540°F.
  - Containment pressure is 4.5 psig.
  - All SG NR levels are <10% with 200 gpm AFW flow to each SG.

Which one of the following completes the following statement?

Based on the current conditions, ECCS flow __(1)__ be reduced because __(2)__.

A. (1) can

(2) all termination criteria have been satisfied

B.✓ (1) can NOT

(2) pressurizer level and RCS subcooling do not meet termination criteria

C. (1) can NOT

(2) RCS subcooling and secondary heat sink do not meet termination criteria

D. (1) can NOT

(2) pressurizer level is the only termination criterion not met

K/A

WE02 SI Termination

EA1.03 Ability to operate and / or monitor the following as they apply to the (SI Termination):

- Desired operating results during abnormal and emergency situations.

K/A MATCH ANALYSIS

The question test the candidate's ability to monitor current plant conditions and determine if SI Termination criteria are met, and if not, which condition(s) is(are) not met.

EXPLANATION OF REQUIRED KNOWLEDGE

EOP 19010-C step 11 checks to see if ECCS flow should be reduced. The following criteria must be met:

- RCS Subcooling GREATER THAN 24 F [38 F ADVERSE].
- Total feed flow to intact SG(s) GREATER THAN 570 GPM.

  -OR-
- NR level in at least one intact SG GREATER THAN 10% [32% ADVERSE].
- RCS pressure STABLE OR RISING.
- PRZR level GREATER THAN 9% [37% ADVERSE].

If all these criteria are met, a transition to 19011-C will be made and ECCS flow will be reduced.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Both pressurizer level and RCS subcooling (32.4°F) are below the required values. However, if the candidate failed to recognize containment pressure is above adverse value then all termination criteria would appear to be met and ECCS flow could be reduced.

B. Correct. ECCS flow cannot be reduced. Both pressurizer level and RCS subcooling are below the required values. Since containment pressure is >3.8 psig, adverse values must be used. Pressurizer level is required to be 37% and based on the stem is currently 30%. RCS subcooling is required to be 38°F and based on stem conditions is calculated to be 32.4°F. RCS pressure conditions and secondary heat sink are within termination limits.

C. Incorrect. Plausible. ECCS flow cannot be reduced. Both pressurizer level and RCS subcooling are below the required values. Since containment

pressure is >3.8 psig, adverse values must be used. Pressurizer level is required to be 37% and based on the stem is currently 30%. RCS subcooling is required to be 38°F and based on stem conditions is calculated to be 32.4°F. RCS pressure conditions and secondary heat sink are within termination limits. However, if the candidate remembers the pressurizer level value incorrectly and mis-reads the AFW flow as 200 gpm total instead of 200 gpm per SG, then the candidate could believe termination criteria are not met on heat sink and RCS subcooling.

D. Incorrect. Plausible. ECCS flow cannot be reduced. Both pressurizer level and RCS subcooling are below the required values. Since containment pressure is >3.8 psig, adverse values must be used. Pressurizer level is required to be 37% and based on the stem is currently 30%. RCS subcooling is required to be 38°F and based on stem conditions is calculated to be 32.4°F. RCS pressure conditions and secondary heat sink are within termination limits. However, if the candidate mis-calculates subcooling by using WR THot insted of CETs, the subcooling would be calculated as 42.4°F and only pressurizer level would not be met.

Level:	RO	
Tier # / Group #	T1 / G2	
K/A#	WE02EA1.03	
Importance Rating:	3.8 / 4.0	
Technical Reference:	19011-C, Rev. 34.3, page 9 Properties of Saturated and Superheated Steam, 2000 ASME STEAM TABLES	
References provided:	Properties of Saturated and Superheated Steam, 2000 ASME STEAM TABLES	
Learning Objective:	LO-LP-37311-08	State how termination/reduced ECCS flow is accomplished. State any differences in the methods used in 19030-C with the directions provided in 19011-C and 19012-C.
	LO-LP-37111-08	Using 19010-C as a guide, briefly describe how each step is accomplished.
	LO-TA-37015	Perform the Initial Recovery Actions for a small Loss of Reactor or Secondary Coolant per 19010-C
	LO-TA-37009	Respond to a Large Break Loss of Primary Coolant per 19010-C
Question origin:	MODIFIED - HL18 # WE02EA2.1	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.8 / 41.10 / 45.5 / 45.6	
Comments:		

You have completed the test!

1. WE02EA2.1 001/1/2/SI TERM - FUNCTIONS/C/A - 3.3/4.2/BANK-FARLEY/HL-18 NRC/RO/SRO/AML

The following conditions exist on Unit 1:

Original Question

- A LOCA is in progress.
- Main Steam Line Isolation has occurred due to Containment pressure.
- 19010-C, "Loss of Reactor or Secondary Coolant," is in progress.

The crew is at the step to, "Check if ECCS flow should be reduced," with plant parameters as follows:

- RCS pressure is 1725 psig and stable.
- CETCs indicate 570°F.
- Total available AFW flow is 580 gpm.
- SG NR levels are all between 12 - 15%.
- PZR level is 30% and slowly rising.

Based on the current conditions, which ONE of the following actions are the operators required to take?

A✓ Continue in 19010-C.

B. Transition to 19011-C, "SI Termination."

C. Transition to 19012-C, "Post-LOCA Cooldown and Depressurization."

D. Transition to 19231-C, "Response to Loss of Secondary Heat Sink."

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***11. Check if ECCS flow should be reduced:**	
a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE].	a. Go to Step 12.
b. Secondary Heat Sink: Total feed flow to intact SG(s) - GREATER THAN 570 GPM. -OR- NR level in at least one intact SG - GREATER THAN 10% [32% ADVERSE].	b. Go to Step 12.
c. RCS pressure - STABLE <u>OR</u> RISING.	c. Go to Step 12.
d. PRZR level - GREATER THAN 9% [37% ADVERSE].	d. Try to stabilize RCS pressure: • Use Normal PRZR Spray if Instrument Air to Containment available. • Do <u>NOT</u> use PRZR PORVs to stabilize RCS pressure. Go to Step 12.
e. Go to 19011-C, ES-1.1 SI TERMINATION.	
***12. Check if Containment Spray should be stopped:**	
a. CS Pumps - RUNNING.	a. Go to Step 13.

Step 12 continued on next page

1. WE04EK1.03 001/LOIT/RO/M/F 3.5/3.9/WE04EK1.03/LO-TA-37020A//HL18 NRC/WE04EK2.01

Initial condition:

- The crew is performing 19112-C, "LOCA Outside Containment."

Current condition:

- RCS pressure is 1500 psig.

Which one of the following completes the following statement?

The FIRST system to be isolated from the RCS to attempt leak isolation is __(1)__,

and

the instrument that will be used to determine isolation of the leak is __(2)__.

	__(1)__	__(2)__
A.	SI	pressurizer pressure
B.	SI	RCS WR pressure
C.	RHR	pressurizer pressure
D.✓	RHR	RCS WR pressure

K/A

WE04 **LOCA Outside Containment**

EK1.03 **Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment):**

- Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment

K/A MATCH ANALYSIS

The question requires the candidate to recall the operational implications associated with the order in which leakage is systematically located and isolated during a LOCA outside containment. The candidate must also recall the associated indications and conditions of successfully isolating the leak during remedial actions.

EXPLANATION OF REQUIRED KNOWLEDGE

Per step 2 of 19112-C, RHR Hot Leg suction and discharge and SIP Hot Leg discharge valves are verified CLOSED in that order. These valves are expected to be CLOSED and therefore no action is required. However, direction is given to CLOSE the valves if

necessary. Step 3 CLOSES RHR Cold Leg discharge valves one at a time. Step 4 CLOSES SI Cold Leg discharge valves one at a time. Therefore, whether the candidate considers the question to address the Hot Leg checks or the Cold Leg manipulations as the FIRST action, RHR is always first.

Following each of the isolation steps, RCS pressure is checked. Rising RCS pressure indicates the leak is isolated. RCS WR pressure is utilized since Pressurizer Pressure is off-scale low at 1700 psig. With plant pressure below 1700 psig, Pressurizer Pressure will continue to indicate 1700 psig.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. The RHR valves are the first to be checked. The SI valves are checked next.

The second part is incorrect. RCS WR pressure is checked to determine if the leak is isolated. Pressurizer pressure is offscale low and indicating 1700 psig at the current plant conditions. However, pressurizer pressure is normally monitored to determine RCS Pressure.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is correct. RCS WR pressure is monitored to determine if the leakage is isolated as described in the Explanation of Required Knowledge above.

C. Incorrect. Plausible. The first part is correct. The RHR valves are the first to be checked or operated as described in the Explanation of Required Knowledge above.

The second part is incorrect. See the second part of choice A above.

D. Correct. The RHR pump valves are the first to be checked or operated as described in the Explanation of Required Knowledge above. Addtionally, RCS WR pressure is monitored to determine if the leak is isolated.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	WE04EK1.03
Importance Rating:	3.5 / 3.9

Technical Reference: EOP 19112-C, Rev 6.2, pages 2 thru 5

References provided: None

Learning Objective:

LO-PP-37116-02 Describe the steps taken to isolate a LOCA outside containment.
LO-PP-37116-03 Describe the indications used to confirm that a LOCA outside containment was successfully isolated.
LO-TA-37020A Respond to a LOCA Outside Contament per 19112-C.

Question origin: BANK - Direct Reuse of HL18 NRC Question # WE04EK2.01

Cognitive Level: M/F

10 CFR Part 55 Content: 41.8 / 41.10 / 45.3

Comments:

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Verify SI reset.	1. IF SI will NOT reset, , THEN initiate ATTACHMENT A.
2. Verify proper valve alignment:	
a. RHR Pump suction from RCS - CLOSED: • HV-8701A - RHR PMP-A DOWNSTREAM SUCTION FROM HOT LEG LOOP-1 • HV-8701B - RHR PMP-A UPSTREAM SUCTION FROM HOT LEG LOOP-1 • HV-8702A - RHR PMP-B DOWNSTREAM SUCTION FROM HOT LEG LOOP-4 • HV-8702B - RHR PMP-B UPSTREAM SUCTION FROM HOT LEG LOOP-4	
b. RHR Pump Hot Leg injection valve - CLOSED: • HV-8840 - RHR TO HL ISO VLV	b. Dispatch an Operator to close affected Unit valve: 1-HV-8840 - RHR TO HL ISO VLV (AB-A13) 2-HV-8840 - RHR TO HL ISO VLV (AB-A18)

° Step 2 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. SI Pump Hot Leg injection valves - CLOSED: • HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV • HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV	c. Dispatch an Operator to close affected Unit valves: 1-HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV (AB-A09) 1-HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV (FHB-A10) 2-HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV (AB-A103) 2-HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV (FHB-A01)
3. Try to identify and isolate RHR Cold Leg injection break:	
a. Close RHR PMP-A TO COLD LEG 1&2 ISO VLV HV-8809A.	
b. Check RCS pressure – RISING.	b. Open RHR PMP-A TO COLD LEG 1&2 ISO VLV HV-8809A. Go to Step 3.d.
c. Go to Step 3.f.	
d. Close RHR PMP-B TO COLD LEG 3&4 ISO VLV HV-8809B.	
e. Check RCS pressure – RISING.	e. Open RHR PMP-B TO COLD LEG 3&4 ISO VLV HV-8809B. Go to Step 4.

Step 3 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
f. Stop RHR Pump in train with leak isolated.	
g. Go to Step 5.	
4. Try to identify and isolate SI Cold Leg injection break:	
a. Close SI PMP-A TO COLD LEG ISO VLV HV-8821A.	
b. Check RCS pressure – RISING.	b. Open SI PMP-A TO COLD LEG ISO VLV HV-8821A. Go to Step 4.d.
c. Go to Step 4.k.	
d. Close SI PMP-B TO COLD LEG ISO VLV HV-8821B.	
e. Check RCS pressure – RISING.	e. Open SI PMP-B TO COLD LEG ISO VLV HV-8821B. Go to Step 4.g.
f. Go to Step 4.k.	
g. Close COLD LEG INJECTION FROM SIS HV-8835.	
h. Check RCS pressure – RISING.	h. Open COLD LEG INJECTION FROM SIS HV-8835. Go to Step 5.
i. Stop both SI pumps.	
j. Go to Step 5.	

Step 4 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
k. Stop SI Pump in train with leak isolated.	
5. Check if break is isolated:	
a. Check RCS pressure – RISING.	a. Go to 19111-C, ECA-1.1 LOSS OF EMERGENCY COOLANT RECIRCULATION.
b. Go to 19010-C, E-1 LOSS OF REACTOR OR SECONDARY COOLANT.	

° END OF PROCEDURE TEXT

1. WE05EA1.02 001/LOIT AND LOCT/RO/C/A 3.7/4.0/WE05EA1.02/LO-TA-37051//HL-18 NRC/054AA1.04

Initial conditions:

- Unit 1 reactor tripped.
- 19231-C, "Response to Loss of Secondary Heat Sink," is in use.
- Bleed and Feed has been initiated.
- 1AA02 is faulted.
- SIP 'B' is running.
- 1PORV-455 is CLOSED.
- 1PORV-456 is OPEN.

Current conditions:

- ALB07-C06 CHARGING PUMP OVERLOAD TRIP is received.
- ALB07-B06 CHARGING LINE HI/LO FLOW is received.
- CCP 'B' handswitch green and amber lights are LIT.

Which one of the following completes the following statement?

Per 19231-C, the minimum requirement for the RCS Feed path __(1)__ met,

and

the Reactor Head vents must be __(2)__ to prevent core uncovery.

	__(1)__	__(2)__
A.✓	is	opened
B.	is	closed
C.	is NOT	opened
D.	is NOT	closed

K/A

WE05 **Loss of Secondary Heat Sink**

EA1.02 **Ability to operate and / or monitor the following as they apply to the (Loss of Secondary Heat Sink):**

- Operating behavior characteristics of the facility

K/A MATCH ANALYSIS

The question tests the candidates ability to monitor plant conditions to determine if adequate feed flow exists. Additionally, the question requires the candidate to relate

these plant conditions to the need to initiate feed and bleed and open head vents when CCPs and both PORVs are not available respectively.

EXPLANATION OF REQUIRED KNOWLEDGE

During a Loss of Secondary Heat Sink event, RCS bleed and feed must be initiated if 3 of 4 SG WR levels lower to les than 29%. This ensures that RCS pressure will be lowered sufficiently to ensure adequate ECCS injection flow is maintained. The PORVs at Vogtle are insufficiently sized to maintain this state indefinitely. If prompt operator action does not occur, RCS pressures will rise to near or above the shutoff head of the CCPs. Eventually, a Loss of Core Cooling (and subsquent core uncovery) will occur.

The event is worsened if a CCP or either PORV is not available. In the event both CCPs are lost, an immediate transition to bleed and feed is required to ensure RCS pressure is lowered below the shutoff head of the SIPs. If performed in a timely manner, the SIPs are an adequate feed source. If either PORV is unavailable, Reactor Head vents are opened to increase the bleed flow rate. However, one PORV and all head vents are not an adequate bleed path and introduction of low pressure water sources to a SG must be initiated immediately due to the increased risk for a Loss of Core Cooling and the reduced time available to implement this contingency action.

Per 19231-C the procedural flow path for the listed plant conditions is as follows:

Step 5	- determine CCPs not available. Stop all RCPs and go to step 33.
Step 34	- actuate SI, which will ensure all available ECCS pumps are running. SIP B is available.
Step 36 & 37	- establish a bleed path through block valves and PORVS. OPEN Rx Head vents if either PORV is not available.
Step 40	- maintain RCS heat removal with SIP flow and PORVs and/or Rx Head vents

ANSWER / DISTRACTOR ANALYSIS

A. Correct.	The first part is correct. Per the current plant conditions, only SIP B is available. With NO CCPs available, bleed and feed must be initiate immediately to reduce RCS pressure below SIP shutoff head. This action establishes an adequate feed source as described in the procedure actions listed in the Explanation of Required Knowledge above. The second part is correct. Since both PORVs could not be OPENED, the RNO of step 37 will OPEN all the Rx Head vents.
B. Incorrect. Plausible.	The first part is correct. See the first part of choice A above. The second part is incorrect. Since both PORVs could not be OPENED, the RNO of step 37 will OPEN all the Rx Head vents. However, when injection flow is lost in 19100-C, Rx Head vents are verified CLOSED to maximize the time to core uncovery. Additionally, the candidate may not recognize (1) PORV is an inadequate bleed path and believe Rx Head vents should remain CLOSED.

C. Incorrect. Plausible. The first part is incorrect. Per the current plant conditions, only SIP B is available. With NO CCPs available, bleed and feed must be initiate immediately to reduce RCS pressure below SIP shutoff head. This action establishes an adequate feed source as described in the procedure actions listed in the Explanation of Required Knowledge above. However, the candidate may believe that, since bleed and feed is immediately initiated with NO CCPs available, a single SIP alone would not constitute an adequate feed source.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G1	
K/A#	WE05EA1.02	
Importance Rating:	3.7 / 4.0	
Technical Reference:	WOG FR-H.1 Background Rev 2, 4/30/2005 WOG ECA 0.0 Background Rev 2, 4/30/2005, page 88 EOP 19100-C, Rev 38.1, pages 6 & 7 EOP 19231-C, Rev 34.0, pages 5 & 20 thru 22	
References provided:	None	
Learning Objective:	LO-LP-37051-11	Define loss of secondary heat sink in accordance with 19231-C, "Response to Loss of Secondary Heat Sink," requiring immediate initiation of bleed and feed control.
	LO-LP-37051-08	Using 19231-C as a guide, briefly describe how each major step is accomplished. Describe the bases for each.
	LO-TA-37051	Respond to a Loss of Secondary Heat Sink per 19231-C.
Question origin:	MODIFIED - HL-18 NRC Question # 054AA1.04, HL17 NRC Question # WE05EK2.1	
Cognitive Level:	C/A	
10 CFR Part 55 Content:	41.7 / 41.10 / 45.5 / 45.6	
Comments:		

You have completed the test!

1. 054AA1.04 001/1/1/LOSS MFW - HPI/C/A - 4.4/4.5/MOD - HL17 NRC/HL-18 NRC/RO/SRO/KAJ

Initial conditions:

- 19231-C, "Response to Loss of Secondary Heat Sink", is in use.
- Bleed and Feed has been initiated.
- SIPs are NOT available.
- Both CCPs are running.
- PORV-455 is CLOSED.
- PORV-456 is OPEN.

HL18 Original Question

Current conditions:

- ALB07-C06, CHARGING PUMP OVERLOAD TRIP illuminates.
- 'B' CCP handswitch green and amber lights are LIT.

Based on the current conditions, which one of the following completes the following statement?

Per 19231-C, the minimum requirement for the RCS Feed path ___(1)___ met,

and

the minimum requirement for the RCS Bleed path ___(2)___ met.

	__(1)__	__(2)__
A.	is	is
B.✓	is	is NOT
C.	is NOT	is
D.	is NOT	is NOT

1. WE05EK2.1 001/1/1/LOHS - COMPONENTS/H-3.7/3.9/NEW/H-17 NRC/RO/SRO/TNT/GCW

Initial conditions:

- 19231-C, "FR-H.1 Response to Loss of Secondary Heat Sink" is in progress.
- RCS Bleed and Feed has been initiated.

Current conditions:

HL17 Original Question

- One SIP is running.
- The CCPs and other SIP are NOT available.

- One PRZR PORV is open.
- The other PORV is NOT available.

Per 19231-C, which one of the following correctly completes the following statement?

One SIP running is ___(1)___ for the RCS Feed path

and

one PRZR PORV open is ___(2)___ for the RCS Bleed path.

For the purposes of this question, "adequate" as defined by procedure 19231-C means that 19231-C will NOT direct further adjustments to the RCS Feed or Bleed path.

	__(1)__	__(2)__
A.	adequate	adequate
B.✓	adequate	NOT adequate
C.	NOT adequate	adequate
D.	NOT adequate	NOT adequate

IMMEDIATE OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

CSFSTs should be monitored for information only. Function Restoration Procedures (FRP) should not be implemented.

1. Verify Reactor trip:
   - Reactor Trip and Bypass Breakers - OPEN.
   - Neutron Flux - LOWERING.

   RESPONSE NOT OBTAINED:

   1. Trip Reactor using both Reactor trip handswitches.

      IF Reactor NOT tripped,
      THEN dispatch an Operator to locally trip the Reactor Trip and Bypass Breakers.

2. Verify Turbine trip:
   - All Turbine Stop Valves - CLOSED.

   RESPONSE NOT OBTAINED:

   2. Trip Turbine.

      IF Turbine will NOT trip,
      THEN run back Turbine.

      IF Turbine can NOT be run back,
      THEN close Main Steamline Isolation and Bypass Valves.

SUBSEQUENT OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. Check if RCS is isolated:

   a. PRZR PORVs - CLOSED.

   RESPONSE NOT OBTAINED:

   a. IF PRZR pressure is less than 2315 psig,
      THEN verify closed affected PRZR PORV(s).

      Perform the following to isolate affected PORV as necessary:

      Open affected PORV power supply breaker:

      AD1M-04 (PV-455A)

      BD1M-04 (PV-456A)

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Letdown Orifice Isolation Valves - CLOSED: • HV-8149A - LETDOWN ORIFICE 45 GPM • HV-8149B - LETDOWN ORIFICE 75 GPM • HV-8149C - LETDOWN ORIFICE 75 GPM	b. Close valves.
c. Letdown Isolation Valves - CLOSED: • LV-0459 - LETDOWN ISOLATION VLV DOWNSTREAM • LV-0460 - LETDOWN ISOLATION VLV UPSTREAM	c. Close valves.
d. Excess Letdown Isolation Valves - CLOSED: • HV-8153 - EXCESS LETDOWN LINE ISO VLV • HV-8154 - EXCESS LETDOWN LINE ISO VLV	d. Close valves.
e. Reactor Vessel Head Vent Isolation Valves - CLOSED: • HV-8095A - RX HEAD VENT TO LETDOWN ISOLATION VLV • HV-8095B - RX HEAD VENT TO LETDOWN ISOLATION VLV • HV-8096A - RX HEAD VENT TO LETDOWN ISOLATION VLV • HV-8096B - RX HEAD VENT TO LETDOWN ISOLATION VLV	e. Close valves.

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
*5. Check CCP status - AT LEAST ONE AVAILABLE.	NO	*5. Stop all RCPs. Go to Step 33.
*6. Check if RCS bleed and feed is required:		
a. Check the following: WR level in any 3 SGs - LESS THAN 29% [44% ADVERSE]. -OR- RCS pressure - GREATER THAN 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK		a. WHEN criteria for bleed and feed are met, THEN perform Step 6.b and Step 6.c. Go to Step 7.
b. Trip all RCPs.		
c. Go to Step 33 and perform bleed and feed actions.		
7. Place Containment Hydrogen Monitors in service by initiating 13130 Post-Accident Hydrogen Control.		
*8. Check CST level - GREATER THAN 15%.		*8. Swap to alternate CST by initiating 13610 Auxiliary Feedwater System.
9. Verify SG Blowdown isolated: • SG Blowdown Isolation Valves - CLOSED WITH HANDSWITCHES IN CLOSE POSITION. • SG Sample Isolation Valves - CLOSED.		

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32. Check for loss of secondary heat sink: WR level in any 3 SGs - LESS THAN 29% [44% ADVERSE]. -OR- RCS pressure - GREATER THAN 2335 PSIG DUE TO LOSS OF SECONDARY HEAT SINK.	32. Return to Step 4.
33. Initiate CONTINUOUS ACTIONS PAGE FOR AFTER ESTABLISHING BLEED AND FEED.	

CAUTION

Step 34 thru Step 37 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

34. Verify SI actuated.

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
35. Verify RCS feed path:	35. Start pumps and align valves as necessary to establish injection flow using Attachment 1 or Attachment 2. IF a feed path can NOT be established, THEN continue attempts to establish feed flow. Return to Step 10.
a. Verify ECCS Pump status: CCPs - AT LEAST ONE RUNNING. -OR- SI Pumps - AT LEAST ONE RUNNING.	
b. Verify ECCS valve alignment - PROPER INJECTION LINEUP INDICATED ON MLBs.	

CAUTION

During bleed and feed operation the PRT may rupture.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
36. Establish RCS bleed path:	
a. Place all PRZR Heaters in OFF/PTL.	
b. Check power to PRZR PORV Block Valves - AVAILABLE.	b. Restore power to block valves.
c. Arm COPS and check PRZR PORV Block Valves - BOTH OPEN.	c. Open both PRZR PORV Block Valves.
d. Open both PRZR PORVs.	

SUBSEQUENT OPERATOR ACTIONS (continued)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
37. Verify adequate RCS bleed path:	37. Perform the following:
• COPS - ARMED.	a. Open Reactor Vessel Head Vent Valves:
• PRZR PORV Block Valves - BOTH OPEN.	• HV-8095A - RX HEAD VENT TO LETDOWN ISOLATION VLV
• PRZR PORVs - BOTH OPEN. NO	• HV-8095B - RX HEAD VENT TO LETDOWN ISOLATION VLV
	• HV-8096A - RX HEAD VENT TO LETDOWN ISOLATION VLV
	• HV-8096B RX HEAD VENT TO LETDOWN ISOLATION VLV
	• HV-0442A - REACTOR HEAD VENT TO PRT
	• HV-0442B - REACTOR HEAD VENT TO PRT
	b. Align an available low pressure water source to at least one intact SG by initiating Attachment 3.
38. Initiate Attachment 4 while continuing with this procedure.	
39. Initiate CONTINUOUS ACTIONS AFTER ESTABLISHING BLEED AND FEED.	
*40. Maintain RCS heat removal:	
• ECCS flow.	
• PRZR PORVs - BOTH OPEN.	• Maintain Reactor Vessel Head Vent Valves open.

STEP: Check If RCS Is Isolated

PURPOSE: To ensure all RCS outflow paths are isolated

BASIS:

A check for RCS isolation is performed to ensure that RCS inventory loss is minimized. The valves itemized are those in major RCS outflow lines that could contribute to rapid depletion of RCS inventory. This step is written for plants which utilize air operated valves (AOVs) in the itemized locations. The step structure assumes that the AOVs fail closed on loss of all ac power (i.e., loss of air supply). The operator, therefore, checks that the valves are closed. If any valve is open, the operator should attempt to close the valve. Reasons for a valve remaining open are plant specific, for example the valves may have legitimate or spurious open signals and air pressure could be available due to air receivers or air bottles located in the air supply system. Plants with air receivers may take up to 30 minutes to lose air pressure. If nitrogen bottles are provided for specific valves, such as PORVs, pneumatic pressure may be available for more than 30 minutes.

The sequence for checking valves is based on capacity of the outflow lines and potential for RCS inventory loss:

1) The pressurizer PORVs are checked first. Since the turbine-driven AFW pump should be running, the secondary side is removing decay heat and RCS pressure should be under the pressurizer PORV setpoint.

2) The letdown line isolation valves adjacent to the RCS loop are checked second. These valves are normally open and receive a low pressurizer level isolation signal. If these valves, in conjunction with the letdown orifice isolation valves, remain open, a leak path to the pressurizer relief tank (PRT) via the letdown line relief valve may exist. These valves, including the letdown orifice isolation valves, if necessary, should be manually closed as soon as possible to

about 40 lbm/sec (290 gpm), with both trains operating, at an RCS pressure of 2300 psig. Since makeup flow from the charging/SI pump system will not keep up with inventory lost out of the pressurizer PORVs, the RCS will eventually dry out enough to cause core uncovery.

In summary, the loss of all feedwater transient from a power condition without operator action will lead to a loss of secondary heat sink followed by a loss of RCS inventory through the pressurizer PORVs. Core uncovery will result at an RCS pressure equal to or greater than the pressurizer PORV setpoint and charging/SI flow, if manually initiated late in the transient, will not be sufficient to prevent core uncovery.

2.2 RCS Bleed and Feed Heat Removal

For a loss of all secondary heat sink, operator action to establish RCS bleed and feed heat removal can prevent or minimize core uncovery. To establish RCS bleed and feed heat removal the operator must initiate and verify high pressure SI flow to feed subcooled fluid to the RCS and then manually open all pressurizer PORVs to bleed hot reactor coolant out of the RCS. To be certain that the bleed and feed heat removal path will be effective, typically at least two PORVs must open.

The effectiveness of RCS bleed and feed heat removal depends on four basic considerations. These are: 1) the timeliness of operator action to initiate bleed and feed following indications of the symptoms of loss of all secondary heat sink (see subsections 2.2.3 and 2.2.4), 2) the core decay heat at the time of RCS bleed and feed initiation, 3) the capacity of the pressurizer PORVs (i.e., number and size of valves), and 4) the capacity of the high pressure SI system (i.e., number, size, and shutoff head of the high pressure SI pumps). These considerations govern the RCS depressurization, repressurization and pressure stabilization after RCS bleed and feed heat removal is established. The fourth consideration also governs the amount of SI flow delivered to the RCS at any RCS pressure. RCS bleed and feed effectiveness is maximized by a combination of these considerations which maximizes the initial RCS depressurization, minimizes the subsequent RCS repressurization and the pressure

1. WE08EK1.02 001/LOIT/RO/M/F 3.4/4.0/WE08EK1.02/LO-TA-37022///

Initial condition:

- A steam rupture inside containment occurred on Unit 1.

Current conditions:

- RCS cold leg temperature is slowly lowering.
- RCS pressure is slowly lowering.
- 19241-C, "Response to Imminent Pressurized Thermal Shock Condition," is in progress.

Which one of the following completes the following statement?

Per 19241-C major operator actions, the in-progress RCS cooldown __(1)__ required to be stopped,

and

the in-progress RCS depressurization __(2)__ required to be stopped.

	__(1)__	__(2)__
A.	is	is
B.✓	is	is NOT
C.	is NOT	is
D.	is NOT	is NOT

K/A

WE08 **RCS Overcooling - PTS**

EK1.02 **Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock):**

- Normal, abnormal, and emergency operating procedures associated with Pressurized Thermal Shock.

K/A MATCH ANALYSIS

This meets the KA since the question tests the operational implications of 19241-C. The operational implications include having to decrease RCS pressure, stabilizing temperature and pressure, and performing a soak. Other procedures are allowed during soak as long as these parameters are not changed.

EXPLANATION OF REQUIRED KNOWLEDGE

The bases behind this step is to prevent propagation of an existing flaw that could cause a through wall crack at the beltline region of the vessel. The KA does not ask for the reason or bases, but the operational implication (i.e., how we would operate the plant differently or be able to perform other actions while in this condition).

Per 19241-C step 6, if RCS WR Cold Leg temperatures are lowering, then the RNO attempts to stop the RCS cooldown by stopping all steam release, throttling RHR, and controlling feed water addition. Cooling the RCS increases the tensile stress on the vessel inner wall.

Per 19241-C step 25, the RCS is to be depressuirzed to lower RCS subcooling. The goal of this step is to reduce subcooling to the minimum value allowed in the EOPs, which minimizes RCS pressure. Lowering RCS pressure reduces the tensile stress on the vessel inner wall.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. The first part is incorrect. RCS temperature must be stabilized to minimize the tensile stress on the reactor vessel inner wall due to the cooldown. However, it is reasonable to assume that a cooldown to <220F to ARM COPs would be an appropriate action for a cold overpressure event. This action is common in many of the EOPs.

The second part is correct. RCS pressure is required to be lowered to <125psig, or subcooling must be between 24F and 34F, or Przr Lvl >75%. With the steam leak present, RCS temperature will continue to lower and therefore RCS pressure will also continue to lowere due to this continuous action step.

B. Incorrect. Plausible. The first part is incorrect. See the first part of choice A above.

The second part is incorrect. RCS pressure is required to be lowered to <125psig, or subcooling must be between 24F and 34F, or Przr Lvl >75%. With the steam leak present, RCS temperature will continue to lower and therefore RCS pressure will also continue to lowere due to this continuous action step. However, in most EOPs, either temperature or pressure is changed at any one time. It is reasonable for a candidate to believe that if RCS temperature is being lowered, RCS pressure will be held stable.

C. Correct. The first part is correct. Per step 6, the cooldown will be stopped (if possible) by isolating all steam (and feed) flow from (and to) the SG's.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is correct. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G2	
K/A#	WE08EK1.02	
Importance Rating:	3.4 / 4.0	
Technical Reference:	19241-C, Rev 25.3, pages 4, 16, & 17	
References provided:	None	
Learning Objective:	LO-LP-37071-04	State the actions for preventing or mitigating the severity of overcooling and repressurizing transients.
	LO-TA-37022	Respond to Imminent Pressurized Thermal Shock per 19241-C.
Question origin:	BANK	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.8 / 41.10 / 45.3	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. Check if RCS cooldown has stopped:	
a. RCS WR Cold Leg temperatures - STABLE OR RISING.	a. Try to stop RCS cooldown: 1) Verify SG ARVs closed. 2) Verify Main Steamline Isolation and Bypass Valves closed. 3) IF RHR system in service, THEN stop any cooldown from RHR system. 4) Stop RCS cooldown to non-faulted SGs by performing the following: • Control feed flow. • Maintain total feed flow greater than 570 gpm until NR level greater than 10% [32% ADVERSE] in at least one non-faulted SG.
7. Check SGs secondary pressure boundaries:	
• SG Pressures: Any lowering in an uncontrolled manner. -OR- Any completely depressurized.	a. Go to Step 9.

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

NOTE

The Upper Head region of the vessel may void during RCS depressurization if RCPs are NOT running. This will result in a rapidly rising PRZR level.

CAUTION

RCS depressurization may result in RCP seal ΔP lowering to less than 200 psid. Shutdown of RCPs is required in this case.

25. Depressurize RCS to lower RCS subcooling:

a. Check if ANY of the following conditions are satisfied: → RNO: a. Go to Step 25.c.

RCS subcooling - 24°F to 34°F [38°F to 48°F ADVERSE].

-OR-

PRZR level - GREATER THAN 75% [52% ADVERSE].

-OR-

RCS pressure - LESS THAN 125 PSIG.

b. Go to Step 29.

Step 25 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Check Normal PRZR Spray - AVAILABLE.	c. IF letdown is in service, THEN go to Step 26. IF letdown is NOT in service, THEN use one PRZR PORV by performing the following: 1) Arm one train of COPS and verify PRZR PORV Block Valve - OPEN. 2) Open associated PRZR PORV. 3) Go to Step 27. IF RCS can NOT be depressurized using any PRZR PORV, THEN go to Step 26 even though letdown is NOT in service.
d. Open Normal PRZR Spray Valves.	
e. Go to Step 27.	
26. Establish Auxiliary Spray by performing the following:	
a. Verify PRZR Heaters - OFF.	
b. Verify at least one CCP running.	

Step 26 continued on next page

1. WE11G2.1.25 001/LOIT AND LOCT/RO/C/A 3.9/4.2/WE11G2.1.25/LO-TA-37020//HL-18 AUDIT/WE11EK1.2

Initial conditions:

- Large break LOCA occurred on Unit 1.
- Containment pressure is 23 psig.
- Both Containment Spray (CS) Pumps are running.
- RWST level is 26% and lowering.
- Four Containment Coolers are running in low speed.

Current conditions:

- 19111-C, "Loss of Emergency Coolant Recirculation," is in progress.

- The crew is at the step to "Determine Containment Spray requirements."

Per 19111-C, which of the following is the required operation of the Containment Spray Pumps?

REFERENCE PROVIDED

A. Stop BOTH CS Pumps immediately and do not restart.

B. Continue to allow BOTH CS Pumps to run until RWST level lowers to < 8%, then stop BOTH CS Pumps.

C. Stop BOTH CS Pumps until suctions can be aligned to the Containment Sump, then restart the CS Pumps that were shut down.

D.✓ Stop ONE CS Pump immediately. Stop the remaining CS pump when RWST level lowers to < 8% if unable to realign to the Containment Sump.

K/A

WE11 **Loss of Emergency Coolant Recirc.**

G2.1.25 **Ability to interpret reference materials, such as graphs, curves, tables, etc.**

K/A MATCH ANALYSIS

The candidate is required to assess current plant conditions and determine the number of Containment Spray Pumps (CSPs) required by interpreting the table in 19111-C 'Loss of Emergency Coolant Recirculation' step 7, which is given as a reference.

EXPLANATION OF REQUIRED KNOWLEDGE

19111-C is entered when the ability to recirculate coolant from the emergency sumps through RHR is lost. The major priority of this procedure is to conserve RWST

inventory to maximize the time to RWST depletion and subsequent stopping of all ECCS pumps. To this end, CSPs are stopped based on available Containment Coolers, Containment Pressure,and RWST level.

For the current plant conditions of 26% RWST level, 23 psig containment pressure, and 4 fans running in low speed, (1) one CSP is required to be in operation. Once the required number of CSPs is determined, pumps and discharge valves are operated as necessary.

In continuous action step 8, a check is made to see if alignment for recirculation should be made. If at least (1) one CSP is running with sufficient level in the sumps, the running CSP(s) will be aligned for recirculation. If no CSPs are running, this step is bypassed. The purpose of this step is to conserve RWST inventory.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Plausible. Per the Explanation of Required Knowledge above, (1) one CSP is required to remain in service. However, the number of fans running in low speed is 4, which does not have a direct correlation in the table. It could be reasonable for a candidate to believe that since more than 3 fans are running in low speed, (0) zero CSPs are required and BOTH CSPs should be stopped.

B. Incorrect. Plausible. Per the Explanation of Required Knowledge above, (1) one CSP is required to remain in service. However, it is common for candidate to improperly navigate the table and cross up rows. As such, it is reasonable for the candidate to believe (2) two CSPs are required and not stop either one. When RWST level lowers <8%, both CSPs would be stopped. This operation is consistent with the normal operation of the CS system.

C. Correct. Per the Explanation of Required Knowledge above, (1) one CSP is required to remain in service. When RWST level lowers to <8%, the remain CSP would be stopped.

D. Incorrect. Plausible. Per the Explanation of Required Knowledge above, (1) one CSP is required to remain in service. However, 4 fans are running in low speed , which does not have a direct correlation in the table. As such, it would be reasonable for a candidate to believe that since more than 3 fans are in low speed, (0) zero CSPs are required and BOTH CSPs should be stopped. Step 8 checks for the need for recirculation alignment. Since BOTH CSPs would be stopped, this step would be bypassed. However, the second part would be correct if (1) one or more CSPs were in operation.

Level:	RO
Tier # / Group #	T1 / G1
K/A#	WE11G2.1.25
Importance Rating:	3.9 / 4.2

Technical Reference: EOP 19111-C Rev 33.2, page 7

References provided: EOP 19111-C Rev 33.2, page 7

Learning Objective:	LO-PP-37115-02	Describe the actions taken to conserve RWST inventory for a loss of emergency coolant recirculation.
	LO-PP-37115-06	Discuss the basis for controlling CS pumps and CNMT Cooler Fans during a loss of emergency coolant recirculation.
	LO-PP-37115-10	Describe the actions taken to keep the core covered and protect ECCS equipment when RWST level drops below 8% during a loss of emergency coolant recirculation.
	LO-TA-37020	Respond to a Loss of Emergency Coolant Recirculation Capability per 19111-C.

Question origin: BANK

Cognitive Level: C/A

10 CFR Part 55 Content: 41.10 / 43.5 / 45.12

Comments:

You have completed the test!

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

b. Determine number of CS Pumps required from Table:

RWST LEVEL	CONTAINMENT PRESSURE	FAN COOLERS IN SLOW	SPRAY PUMPS REQUIRED
GREATER THAN 29%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 21.5 PSIG and 52 PSIG	0	2
		4	1
		8	0
	LESS THAN 21.5 PSIG	N/A	0
BETWEEN 8% and 29%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 21.5 PSIG and 52 PSIG	3	1
		6	0
	LESS THAN 21.5 PSIG	N/A	0
LESS THAN 8%	N/A	N/A	0

c. Check CS Pumps running - EQUAL TO NUMBER REQUIRED.

c. Reset Containment Spray.

Operate CS Pumps and discharge valves as required.

°

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

b. Determine number of CS Pumps required from Table:

RWST LEVEL	CONTAINMENT PRESSURE	FAN COOLERS IN SLOW	SPRAY PUMPS REQUIRED
GREATER THAN 29%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 21.5 PSIG and 52 PSIG	0	2
		4	1
		8	0
	LESS THAN 21.5 PSIG	N/A	0
BETWEEN 8% and 29% 26%	GREATER THAN 52 PSIG	N/A	2
	BETWEEN 23 psig 21.5 PSIG and 52 PSIG	3 4	1
		6	0
	LESS THAN 21.5 PSIG	N/A	0
LESS THAN 8%	N/A	N/A	0

c. Check CS Pumps running - EQUAL TO NUMBER REQUIRED.

NO

c. Reset Containment Spray.

Operate CS Pumps and discharge valves as required.

1. WE16EK2.02 001/LOIT/RO/M/F 2.6/3.0/WE16EK2.02/LO-TA-37009/13009///

Given the following:

- Large break LOCA is in progress on Unit 1.
- 1RE-002, 1RE-003, 1RE-005, and 1RE-006 are in high alarm.
- 19013-C, "Transfer to Cold Leg Recirculation," is in progress.
- The crew just completed realigning Containment Spray suctions.

Which one of the following completes the following statement?

Per 19013-C, if dose rates will not allow reading local suction and discharge pressure indications, proper operation of Containment Spray shall be verified by observing containment __(1)__ lowering,

and

per 19010-C, "Loss of Reactor or Secondary Coolant," containment spray is required to remain in the recirculation mode for no less than __(2)__ hours.

	__(1)__	__(2)__
A.✓	pressure	1.5
B.	pressure	2
C.	temperature	1.5
D.	temperature	2

K/A

WE16 **High Containment Radiation**

EK2.02 **Knowledge of the interrelations between the (High Containment Radiation) and the following:**

- Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

K/A MATCH ANALYSIS

Containment Spray is a heat removal system employed during a large break LOCA. Emergency coolant is recirculated via the containment emergency sumps and sprayed

into the containment atmosphere to cool containment and lower overall containment tempertaure and pressure. There is an interrelationship between high containment radiation and expected dose rates at the CS Pumps rooms, which is the only location where suction and discharge pressures can be read. An additional iterrelationship exists between the minimum recirculation time requirements and the entrainment of Iodine inside containment.

NOTE: K/A category WE16 specifically addresses FR-Z.3, "Repsonse to High Containment Radiation Level". The specific knowledge item EK2.02 addresses an interralationship with heat removal. At Vogtle our FR-Z.3 (EOP 19253-C) does not have any direction association with any heat removal system. The closest interrelationship within the EOP network between High Containment Radiation and a Containment Heat Removal system would be Containment Spray on Sump Recirc during a LOCA event with High Containment Radiation. Therefore, this is the approach taken with this question.

EXPLANATION OF REQUIRED KNOWLEDGE

With RE-005 and 006 in high alarm, fuel damage is expected. As such, dose rates in the area of the Containment Spray pump rooms will be too high for personnel to be dispatched to verify suction and discharge pressures, per 19013-C guidance. These indications are only available locally. Therefore, 19013-C has a CAUTION prior to step 17 that states dispatch will only be performed if radiation levels permit. Step 18.c. uses stable or lowering containment pressure as an alternte indication of proper operation.

Per step 12.d. of 19010-C, Containment Spray is required to be operated for 2 hours, 1.5 hours of which is in recirc mode if any of the containment rad monitors (RE-002, 003, 005, or 006) are in high alarm. The 2 hour operation requirement is for iodine scrubbing. During this period, iodine is "washed" out of the containment atmosphere and into solution in the emergency sumps. The 1.5 hours on recirculation is required to ensure proper mixing of the TSP located in baskets on the containment floor into the emergency sump fluid. The TSP raises the PH level of the water in the sump, minimizing corrosion on containment structures and equipment as the sump water is sprayed in containment.

ANSWER / DISTRACTOR ANALYSIS

A. Correct. The first part is correct. Per 19013-C CAUTION above step 17 and step 18.c, if radiation levels do not allow local verification of suction and discharge pressures, stable or lowering containment pressure is sufficient to alternately ensure proper operation.

The second part is correct. Per 19010-C step 12.d, at least 1.5 hours in recirculation mode is required.

B. Incorrect. Plausible. The first part is correct. See the first part of choice A above.

The second part is incorrect. Per 19010-C step 12.d, at least 1.5 hours in recirculation mode is required. However, step 12.d requires the Containment Spray Pumps to operate for a

minimum of 2 hours total for iodine scrubbing.

C. Incorrect. Plausible. The first part is incorrect. Per 19013-C step 18.c, if radiation levels do not allow local verification of suction and discharge pressures, stable or lowering containment pressure is sufficient to alternately ensure proper operation. However, since Containment Spray lowers pressure by lowering the bulk atmosphere temperature, monitoring temperature could be a reasonable response to ensure the CS system is operating properly.

The second part is correct. See the second part of choice A above.

D. Incorrect. Plausible. The first part is incorrect. See the first part of choice C above.

The second part is incorrect. See the second part of choice B above.

Level:	RO	
Tier # / Group #	T1 / G2	
K/A#	WE16EK2.02	
Importance Rating:	2.6 / 3.0	
Technical Reference:	EOP 19010-C, Rev 34.3, pages 9 & 10 EOP 19013-C, Rev 29.2, pages 9 & 10	
References provided:	None	
Learning Objective:	LO-LP-37113-02	Using EOP 19013-C as a guide, briefly describe how each step is accomplished.
	LO-LP-37113-05	Given a NOTE or CAUTION statement from the EOP, state the bases for that NOTE or CAUTION statement.
	LO-LP-37111-08	Using EOP 19010-C as a guide, briefly describe how each step is accomplished.
	LO-TA-13009	Manually align ECCS for Cold Leg Recirculation Phase using EOP 19013-C.
	LO-TA-37009	Respond to a Large Break Loss of Primary Coolant per 19010-C
	LO-PP-15101-06	State the reason for a minimum required time the Containment Spray system is left on recirculation following a LOCA.
Question origin:	NEW	
Cognitive Level:	M/F	
10 CFR Part 55 Content:	41.7 / 41.9 / 41.10 / 45.7	
Comments:		

You have completed the test!

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
***11. Check if ECCS flow should be reduced:**	
a. RCS Subcooling - GREATER THAN 24°F [38°F ADVERSE].	a. Go to Step 12.
b. Secondary Heat Sink: Total feed flow to intact SG(s) - GREATER THAN 570 GPM. -OR- NR level in at least one intact SG - GREATER THAN 10% [32% ADVERSE].	b. Go to Step 12.
c. RCS pressure - STABLE <u>OR</u> RISING.	c. Go to Step 12.
d. PRZR level - GREATER THAN 9% [37% ADVERSE].	d. Try to stabilize RCS pressure: • Use Normal PRZR Spray if Instrument Air to Containment available. • Do <u>NOT</u> use PRZR PORVs to stabilize RCS pressure. Go to Step 12.
e. Go to 19011-C, ES-1.1 SI TERMINATION.	
***12. Check if Containment Spray should be stopped:**	
a. CS Pumps - RUNNING.	a. Go to Step 13.

° Step 12 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Containment pressure - LESS THAN 15 PSIG.	b. WHEN Containment pressure is less than 15 psig, THEN go to Step 12.c. Go to Step 13.
c. Any Containment radiation levels - INDICATE HIGH DUE TO PRIMARY LOCA: RE-002 RE-003 RE-005 RE-006	c. Perform the following: 1) Reset Containment Spray signal. 2) Stop Containment Spray Pumps. 3) Close CNMT SPRAY ISO VLV: • HV-9001A • HV-9001B Go to Step 13.
d. Operate CS Pumps: • Minimum of 2 hours. • At least 1.5 hours in recirculation mode.	d. WHEN CS Pumps have operated for at least 2 hours AND in the recirculation mode for at least 1.5 hours, THEN perform Step 12.c RNO.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- The specified actions in Steps 17 through 19 should be promptly completed to avoid loss of CS Pump suction.
- Local observation of CS Pump suction and discharge pressure gauges should only be performed if radiation levels permit.

  UNIT 1 (AB D75) UNIT 2 (AB D06)

17. Reset Containment Spray.

18. Align CS Pump A for recirculation:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Open CS Pump A suction valves from Containment Emergency Sump: • HV-9002A, CNMT SPRAY PUMP A CNMT SUMP SUCT IRC • HV-9003A, CNMT SPRAY PUMP A CNMT SUMP SUCT ORC	a. Locally open: 1-HV-9003A (AB-C134) 2-HV-9003A (AB-C124) IF valves can NOT be opened, THEN stop CS Pump A. Go to Step 19.
b. Close CNMT SPRAY PUMP A RWST SUCT ISO VLV: • HV-9017A	

Step 18 continued on next page

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. Check Train A CS proper operation using the following indications, if available: Pump suction pressure PI-0972 - GREATER THAN 7 PSIG. Pump discharge pressure PI-0974 - APPROXIMATELY 185 PSIG ABOVE SUCTION PRESSURE. Containment pressure - STABLE OR LOWERING.	c. Verify valve alignment correct: • HV-9002A - OPEN • HV-9003A - OPEN • HV-9017A - CLOSED • HV-9001A - OPEN
19. Align CS Pump B for recirculation:	
a. Open CS Pump B suction valves from Containment Emergency Sump: • HV-9002B, CNMT SPRAY PUMP B CNMT SUMP SUCT IRC • HV-9003B, CNMT SPRAY PUMP B CNMT SUMP SUCT ORC	a. Locally open: 1-HV-9003B (FHB-C08) 2-HV-9003B (FHB-C02) IF valves can NOT be opened, THEN stop CS Pump B. Go to Step 20.
b. Close CNMT SPRAY PUMP B RWST SUCT ISO VLV: • HV-9017B	

Step 19 continued on next page

SUBSEQUENT OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. Verify Containment Ventilation Isolation: a. Dampers and Valves - CLOSED: CVI MLB indication -OR- Reference ATTACHMENT 1 as necessary.	
2. Check Piping Penetration Filtration and Exhaust Units - BOTH RUNNING.	2. Start fans by initiating 13305-1, 13305-2 Auxiliary Building HVAC System.
3. Place the Containment Preaccess Filter units in service by initiating 13125-1, 13125-2 Containment Purge System.	
4. Notify TSC of Containment radiation level to obtain recommended action.	
5. Return to procedure and step in effect.	

END OF PROCEDURE TEXT