Salem December 2014 NRC Exam

RO Question #61

Given the following conditions:

- Unit 2 is operating at 100% power with an identified small fuel pin leak.
- A 5 gpm tube leak occurs on 22 SG.

Of the following, which is the only radiation monitor that will NOT show a change from this tube leak?

a. 2R19B, 22 SG Blowdown.
b. 2R46A, 22 Main Steam Line.
c. 2R15, Condenser Air Ejector.
d. 2R41D, Plant Vent Release Rate.

Correct Answer: b

Proposed Change: Delete question based on no correct answer

Justification: The R46 monitors are designed to provide continuous monitoring of high-level, post-accident releases of radioactive noble gases via the safety-relief valves, atmospheric dump valves, and auxiliary feed pump turbine and are capable of functioning both during and following an accident. The monitors are designed to meet the requirements of NUREG-0737 II.F.1 and the intent of RG 1.97. A range of 1E-1 to 1E3 micro Ci/cc is required. The monitor's alarm function (7 mR/hr) is used in the EOPs to identify a SGTR event/EOP entry point, and identify which SG has ruptured.

However, when discussing the post-accident ability of the monitor, the most recent DCP to update the R46 monitors installed in 2005/2006, DCP 80057587, says on page 22 of 109...."However, these monitors will exhibit a response to N-16, but typically only at levels corresponding to leak rates in the gallon per minute range. ***Because of the N-16 response, these monitors can provide a clear indication of a SGTR if the rupture occurs while the unit is at power.***" (my emphasis)

Since the 2R46A will exhibit a response to the SGTL described in the stem (5 gpm at 100% power), as well as the remaining three distracters, we believe there is no correct answer to this question, and request it be deleted.

gamma rays (such as Xe-133 with an 80 KeV photon) is minimal. These monitors would only respond if there was a significant RCS (Reactor Coolant System) source term. As a result, these monitors cannot be used for low level leak rate detection and are limited to post-accident assessment of significant releases. However, these monitors will exhibit a response to N-16, but typically only at levels corresponding to leak rates in the gallon per minute range. Because of the N-16 response, these monitors can provide a clear indication of a SGTR if the rupture occurs while the unit is at power. However, once the reactor trips the readings will typically return to normal levels.

Because they are not sensitive to small primary-to-secondary leak changes, Main Steam Line Monitors installed to meet RG 1.97 requirements will provide no useful primary-to-secondary leak trend information.

The off-line sampling arrangement currently used is not discussed but would have a considerably lower N-16 response due to sample transport time and the short N-16 half-life of ~7 seconds. The sensitivity of the existing arrangement to low energy gammas is somewhat higher due to less pipe thickness.

'0273

2.10 Existing SAP Classification

The existing R46 Monitors are classified in SAP as EQ Mild; Safety Related; QA Required Yes; Seismic Classification 1; Safety Significance 002 ; Component Classification Data Q07 (Safety Sy M/I/Cont Pen Cmp); Safety Class/Quality Grp Code 1E ; Safety Class (Nuclear) 1E; Tech Spec Accpt Criteria Inst

2.11 Steam Line Penetration Room Environmental Information

Per S-C-ZZ-SDC-1419, *Salem Generating Station Environmental Design Criteria*, the mechanical steam line penetration rooms have the following design-basis-accident environmental conditions: 5.4 PSIG, 467 DEG F, 100% RH, 187 mR/Hr, 5.63 RAD, and 356R TID.

2.12 Design Bases Summary

The R46 monitors are required to provide continuous monitoring of high-level, post-accident releases of radioactive noble gases via the safety-relief valves, atmospheric dump valves, and auxiliary feedpump turbine and are to be capable of functioning both during and following an accident. The monitors are designed to meet the requirements of NUREG-0737 II.F.1 and the intent of RG 1.97. A range of 10-1 to 103 μCi/cc is required although equivalency to Xe-133 not explicitly required by RG 1.97. RG- 1.97 acknowledges that monitors placed adjacent to main steam lines would not be able to detect low energy gamma emissions and a lower energy response of 500 Kev is acceptable, as long as the concentration of lower energy-emitting isotopes can be

Salem December 2014 NRC Exam

SRO Question #6

Given the following conditions:

- Unit 1 is operating at 100% power.
- A breaker fault occurs on the 2-6 500 KV breaker.
- The 2-6 500 KV breaker does NOT trip, but should have.
- 15 seconds after the breaker failure, Unit 1 has NOT tripped.

Which of the following identifies how the Unit 1 CRS should proceed?

a. Direct the RO to manually trip the reactor and go to EOP-TRIP-1, Reactor Trip or Safety Injection. Concurrently with EOP implementation, initiate S1.OP-AB.LOOP-0001, Loss of Offsite Power, and perform Attachment 2, Loss of Group Buses, Part A, Loss of 1E and 1H 4KV Group Buses.
b. Direct the RO to manually trip the reactor and go to EOP-TRIP-1. Concurrently with EOP implementation, initiate S1.OP-AB.LOOP-0001, and perform Attachment 2, Loss of Group Buses, Part B, Loss of 1F and 1G 4KV Group Buses.
c. Enter S1.OP-AB.LOOP-0003, Partial Loss of Off-Site Power, then enter S1.OP-AB.CW-0001 Circulating Water System Malfunction, and perform a power reduction to 83% power or less.
d. Enter S1.OP-AB.LOOP-0003, then enter S1.OP-AB.CW-0001 and open the Hood Spray Bypass valves 11-13MC62s.

Correct Answer: d

Proposed Change: Accept choice c as correct also.

Justification: Choices c and d both have the correct procedure progression, with choice d having an action directed at step 3.12 of AB.CW with 3 circulators running, with one running on each waterbox. Choice c contains the action to reduce power to $\leq$83% power, which would be performed directly IAW step 3.9 if both the 11A & 11B circulators are out of service. Additionally, the Continuous Action Summary, Step 5.0 (page 11 of 32 in AB.CW) states…

<u>"IF AT ANYTIME</u> any of the following conditions exist or are approaching:

- Condenser ΔT >27°F
- Condensate suction temperature $\geq$135°F
- Flashing occurs in the Condenser Hotwell or Condensate Pump suction piping as indicated by erratic Condensate Pump Amp Indication
  OR erratic SGFP suction pressure indication.

THEN **INITIATE** a load reduction IAW S1.OP-AB.LOAD-0001(Q),

Rapid Load Reduction, concurrently with this procedure until parameters stabilize within operational limits."

Based on their knowledge of plant response to a loss of circulators, 4 of 8 SRO candidates assumed that condenser ΔT would rise rapidly and exceed the 27°F limit, and based their answer on having to do a rapid load reduction to some power level below 83%. The accompanying simulator charts reflecting a breaker failure on the 2-6 500KV breaker corroborates that rapid rise in condenser temperature, and the necessity for initiating the power reduction. As shown on the Plant Computer screen shot, Rx power was lowered to approximately 81% to reduce Condenser ΔT to 27.1°F. An additional simulator computer graph shows that a Rx power reduction to 81% power was required to lower condenser ΔT to 27 °F.

ATTACHMENT 1
(Page 2 of 7)

CONTINUOUS ACTION SUMMARY

___ 4.0 IF AT ANYTIME a load reduction is required to maintain condenser backpressure within the Allowable Operating Region of Attachment 4, Condenser Absolute Pressure Limits, THEN:

> **NOTE**
>
> ___ Turbine load reduction ramp rates of $\leq$5%/min are desirable to prevent operation of Steam Dumps, which could degrade the low vacuum/high temperature condition.
>
> ___ The degradation of condenser vacuum and possible loss of Steam Dumps, may lead to the use of the MS10 atmospheric reliefs for temperature control.

- **INITIATE** a Rapid Load Reduction IAW S1.OP-AB.LOAD-0001(Q) concurrently with this procedure until parameters stabilize within the operational limits of Attachment 4. ____ Time

___ 5.0 IF AT ANYTIME any of the following conditions exist or are approaching:

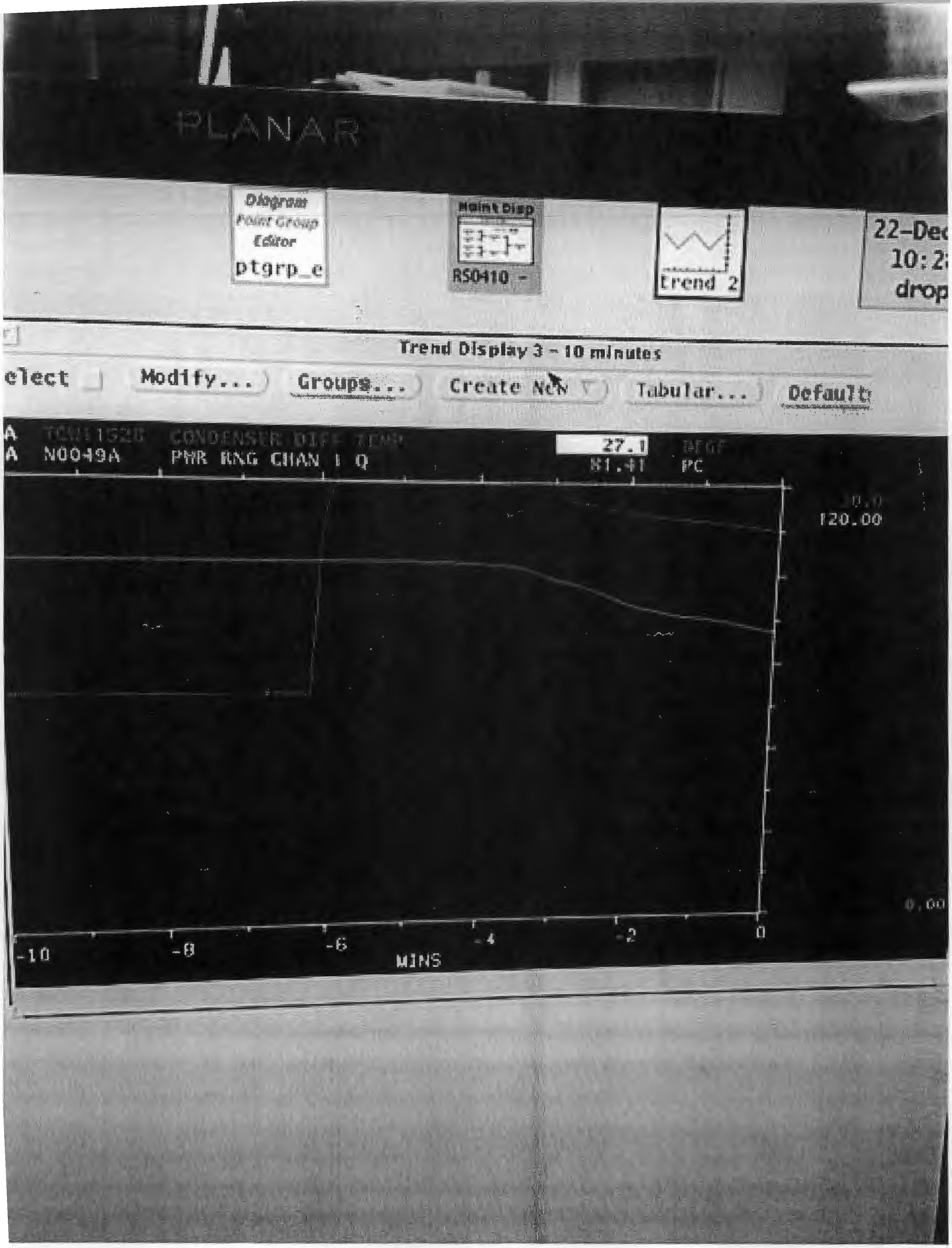
- Condenser ΔT >27°F
- Condensate suction temperature $\geq$135°F
- Flashing occurs in the Condenser Hotwell or Condensate Pump suction piping as indicated by erratic Condensate Pump Amp Indication OR erratic SGFP suction pressure indication.

THEN **INITIATE** a load reduction IAW S1.OP-AB.LOAD-0001(Q), Rapid Load Reduction, concurrently with this procedure until parameters stabilize within operational limits. ____ Time

___ 6.0 IF AT ANYTIME Hotwell level is <32 inches AND there are indications of Condensate Pump cavitation, THEN:

___ 6.1 **STOP** the affected CN Pump.

___ 6.2 **INITIATE** S1.OP-AB.CN-0001(Q), Main Feedwater/Condensate System Abnormality, concurrently with this procedure. ____ Time


PLANAR
Diagram Point Group Editor
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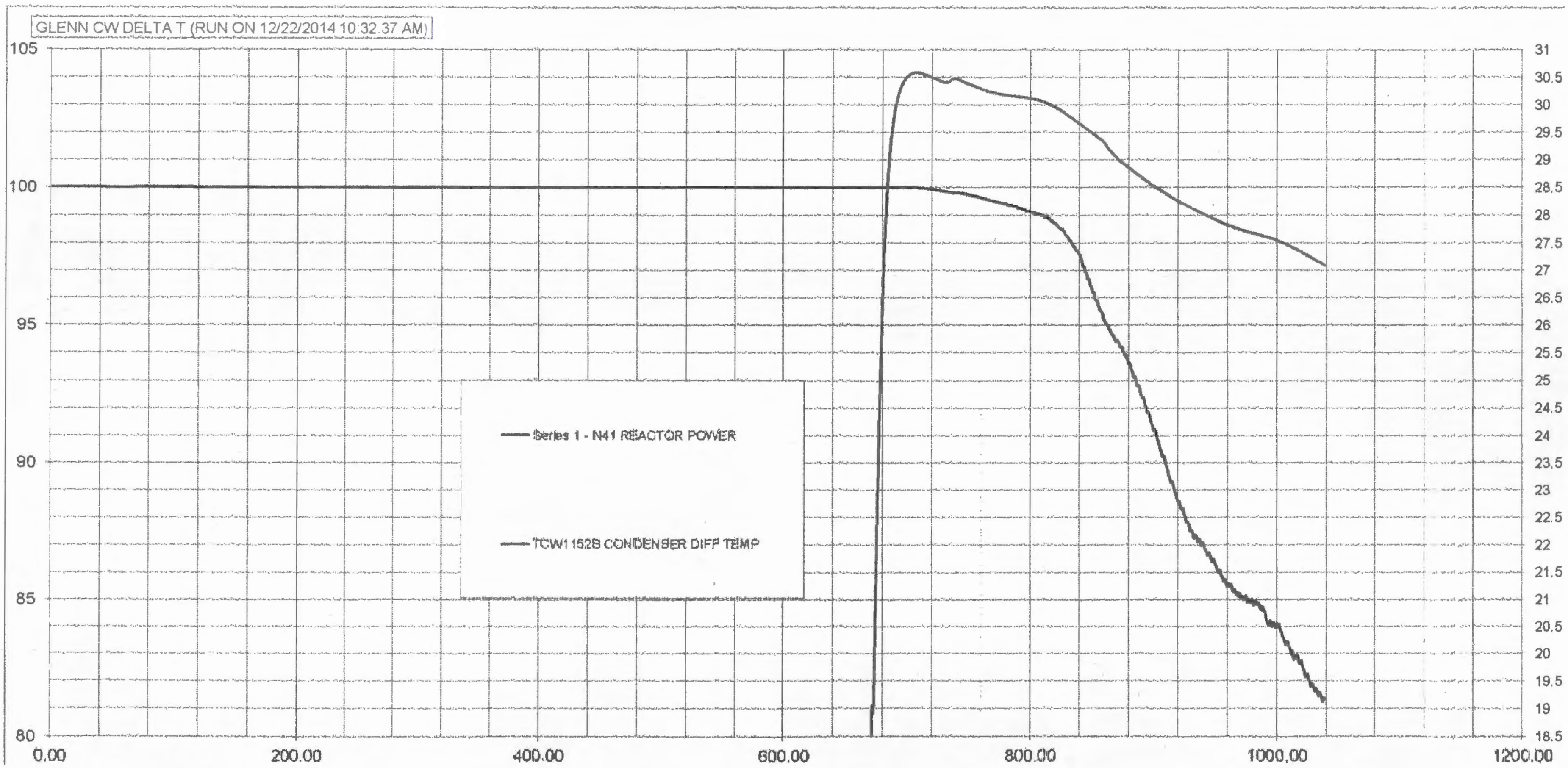
MINS


GLENN CW DELTA T (RUN ON 12/22/2014 10.32.37 AM)
Series 1 - N41 REACTOR POWER
TCW1152B CONDENSER DIFF TEMP
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100
95
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CIRCULATING WATER SYSTEM MALFUNCTION

1.0 ENTRY CONDITIONS

DATE: ______ TIME: ____

1.1 Any indication of abnormal operation of the Circulating Water System.

1.2 Any indication of a system rupture, in the Circulating Water System, external to the condenser.

1.3 Any indication of a gross condenser tube failure.

1.4 Removal of two or more Circulating Water Pumps from service.

2.0 IMMEDIATE ACTIONS

None

3.0 SUBSEQUENT ACTIONS

___ 3.1 **INITIATE** Attachment 1, Continuous Action Summary.

3.2 Is there a pipe rupture in the Circulating Water System as indicated by any of the following? **[C0361]**

- Report from personnel of excessive leakage from the Circulating Water System.
- OHA G-43, TURB AREA LVL HI PMP START

___ NO ___ YES ——> **GO TO** Step 3.20

V

____ Time

3.3 Are two or more Circulating Water Pumps O/S?

___ YES ___ NO ——> **GO TO** Step 3.26

V

____ Time

NOTE
___ Steam Dump operation is blocked to those Condensers which have both Circulating Water Pumps out of service, or CNDSR OUTLET MOVs do <u>NOT</u> indicate open.

3.4 Is at least one Circulating Water Pump operating on each condenser?

___ NO ___ YES ——> **GO TO** Step 3.11

V

_____
Time

___ 3.5 <u>IF</u> 11A and 11B Circulating Water Pumps are BOTH out of service,
<u>THEN</u> **GO TO** Step 3.9

NOTE
___ When 11 - 14GB4 valves are closed then SGBD sodium is no longer a representative parameter of Steam Generator chemistry.

___ 3.6 <u>IF</u> Steam Generator Blowdown is in service to 12 Condenser,
<u>AND</u> 12A and 12B Circulating Water Pumps are BOTH out of service,
<u>THEN</u>

___ A. **CLOSE** 11-14GB4, BLOWDOWN ISOLATION VALVEs.

___ B. **INITIATE** S1.OP-SO.GBD-0002(Q), Steam Generator Blowdown Operation, to complete Steam Generator Blowdown isolation, as required.

___ C. <u>IF</u> 12A and 12B are the ONLY Circulating Water Pumps out of service, <u>THEN</u> **CONTINUE** to monitor CAS limits and evaluate the need to initiate a load reduction.

___ D. **PLACE** SGBD in service to 13 Condenser as directed by the SM/CRS.

___ E. **GO TO** Step 3.8

NOTE

___ When 11 - 14GB4 valves are closed then SGBD sodium is no longer a representative parameter of Steam Generator chemistry.

___ 3.7 IF Steam Generator Blowdown is in service to 13 Condenser, AND 13A and 13B Circulating Water Pumps are BOTH out of service, THEN:

___ A. **CLOSE** 11-14GB4, BLOWDOWN ISOLATION VALVEs.

___ B. **INITIATE** S1.OP-SO.GBD-0002(Q), Steam Generator Blowdown Operation, to complete Steam Generator Blowdown isolation, as required.

___ C. IF 13A and 13B are the ONLY Circulating Water Pumps out of service, THEN **CONTINUE** to monitor CAS limits and evaluate the need to initiate a load reduction.

___ D. **PLACE** SGBD in service to 12 Condenser as directed by the SM/CRS.

___ 3.8 Are conditions stable and CAS limits maintained?

___ NO ___ YES ——> **GO TO** Step 3.11

V

___ Time

NOTE

___ Turbine load reduction ramp rate of 5%/min or less is desirable to prevent operation of Steam Dumps, which could exacerbate the low vacuum/high temperature condition.

___ 3.9 **INITIATE** a Rapid Load Reduction to ≤83% Reactor Power IAW S1.OP-AB.LOAD-0001(Q), Rapid Load Reduction, to prevent flashing in the Condensate System, while continuing with this procedure. **[C0346]**

___ Time

___ 3.10 When Reactor Power is stable at ≤83% due to two Circulating Water Pumps out of service in any waterbox, **EVALUATE** plant conditions AND **PERFORM** one of the following while continuing with this procedure:

- Maintain power stable to maintain CAS limits.
- Lower power level to maintain CAS limits.
- Raise power (if desired) until CAS limits are approached IAW Attachment 6, Raising Power with Two Circulating Water Pumps Out of Service in any Waterbox.

___ 3.11 IF unable to maintain the affected condenser level(s) >33 inches, THEN **SEND** an Operator to locally throttle the applicable (A or B) condenser hotwell isolation valve(s) to maintain between 33 and 65 inches: [C0346]

- ___ ♦ 11A Hotwell 11CN79, 11A CONDENSER HOTWELL ISOL
- ___ ♦ 11B Hotwell 11CN83, 11B CONDENSER HOTWELL ISOL
- ___ ♦ 12A Hotwell 12CN79, 12A CONDENSER HOTWELL ISOL
- ___ ♦ 12B Hotwell 12CN83, 12B CONDENSER HOTWELL ISOL
- ___ ♦ 13A Hotwell 13CN79, 13A CONDENSER HOTWELL ISOL
- ___ ♦ 13B Hotwell 13CN83, 13B CONDENSER HOTWELL ISOL

___ 3.12 **SEND** Operators to:

CAUTION

___ **The use of LP Hood Spray is permissible >15%power as a mitigating strategy for existing conditions. However, the use of LP Hood Spray is not permitted under normal circumstances >15% power because prolonged use at >15% power can result in LP Turbine Blade damage. [70039711]**

- ___ ♦ **OPEN** the appropriate 11MC62, 12MC62, or 13MC62, TURB HOOD SPRAY BYP V, on the affected condenser(s). [C0346]
- ___ ♦ **INITIATE** monitoring of Condenser Hotwell and Condensate Pump suction piping for indications of flashing.
- ___ ♦ **DETERMINE** cause of Circulating Water Pump(s) failure.

ATTACHMENT 1
(Page 2 of 7)

CONTINUOUS ACTION SUMMARY

___ 4.0 IF AT ANYTIME a load reduction is required to maintain condenser backpressure within the Allowable Operating Region of Attachment 4, Condenser Absolute Pressure Limits, THEN:

NOTE
___ Turbine load reduction ramp rates of ≤5%/min are desirable to prevent operation of Steam Dumps, which could degrade the low vacuum/high temperature condition.
___ The degradation of condenser vacuum and possible loss of Steam Dumps, may lead to the use of the MS10 atmospheric reliefs for temperature control.

- ♦ **INITIATE** a Rapid Load Reduction IAW S1.OP-AB.LOAD-0001(Q) concurrently with this procedure until parameters stabilize within the operational limits of Attachment 4. ______ Time

___ 5.0 IF AT ANYTIME any of the following conditions exist or are approaching:

- ♦ Condenser ΔT >27°F
- ♦ Condensate suction temperature ≥135°F
- ♦ Flashing occurs in the Condenser Hotwell or Condensate Pump suction piping as indicated by erratic Condensate Pump Amp Indication OR erratic SGFP suction pressure indication.

THEN **INITIATE** a load reduction IAW S1.OP-AB.LOAD-0001(Q), Rapid Load Reduction, concurrently with this procedure until parameters stabilize within operational limits. ______ Time

___ 6.0 IF AT ANYTIME Hotwell level is <32 inches AND there are indications of Condensate Pump cavitation, THEN:

___ 6.1 **STOP** the affected CN Pump.

___ 6.2 **INITIATE** S1.OP-AB.CN-0001(Q), Main Feedwater/Condensate System Abnormality, concurrently with this procedure. ______ Time

Salem December 2014 NRC Exam

SRO Question #8

Given the following conditions:

- Unit 2 is operating at 100% power.
- Operators receive the following alarms:
- OHA B-13 21 SW HDR PRESS LO
- OHA B-14 22 SW HDR PRESS LO
- SW header pressure indication in the control room reads 98 psig for both headers.

No other OHA's have annunciated.

Which of the following describes both the possible location of a Service Water System leak which would cause these indication, and how the CRS should respond?

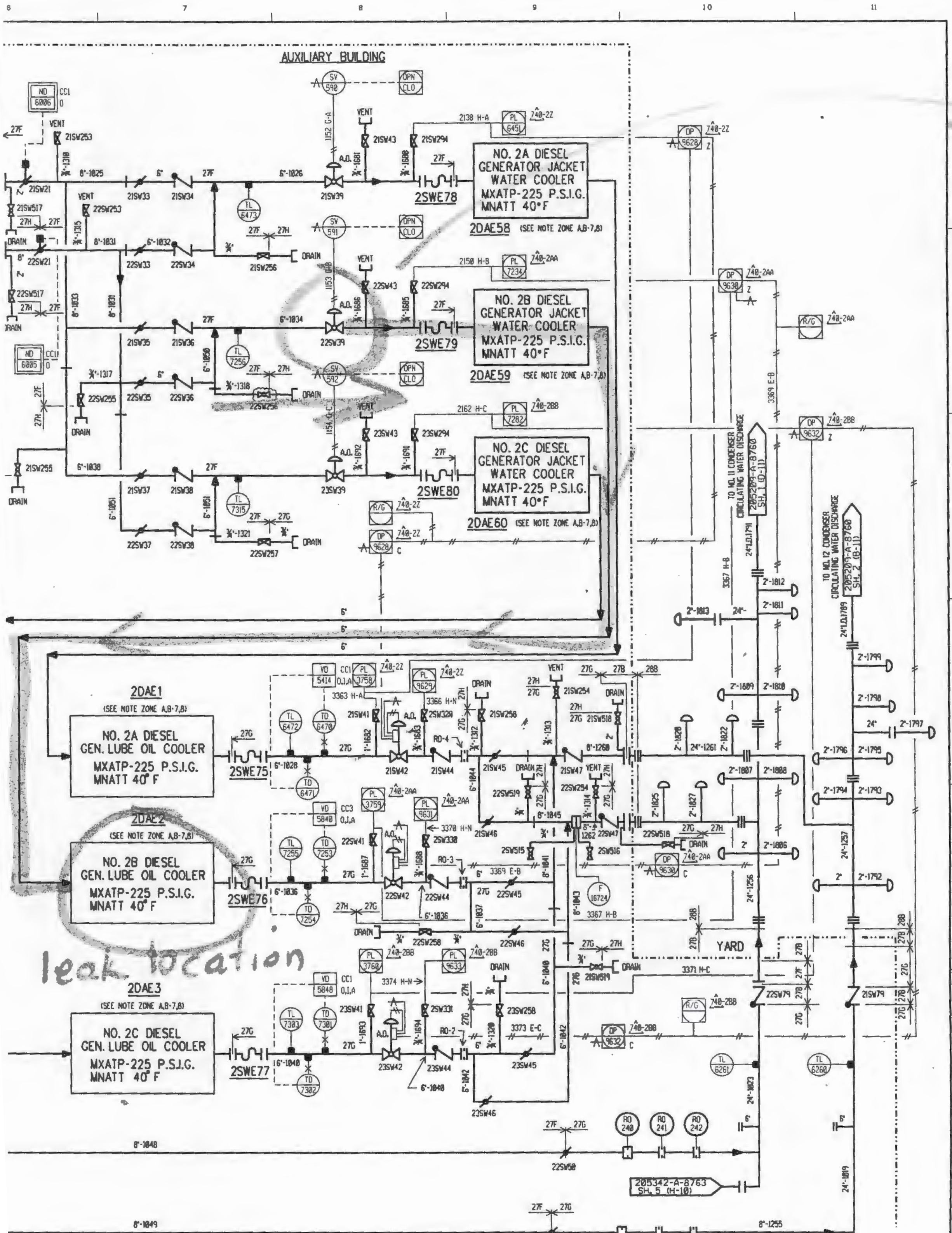
Assume each of the leaks is large enough to cause the indications present in the control room.

a. 4 Service Water Bay. Split SW Bays by closing 21SW17 and 22SW17 IAW S2.OP-AB.SW-0003, Service Water Bay Leak.
b. Nuclear header x-over line between the 21SW23 and the 22SW23. Shut EITHER SW23 using Attachment 6, Service Water Valve Malfunctions, of S2.OP-AB.SW-0001.
c. 2B EDG Lube Oil Cooler. Isolate BOTH SW supply header isolation valves and BOTH return header isolation valves to 2B EDG IAW S2.OP-AB.SW-0001 Loss of Service Water Header Pressure.
d. 21 CCW HX end bell. Ensure 22 CC HX controller set lower than 21 CCW HX, and isolate 21 Service Water Header using Attachment 4, Service Water Header Isolation, of S2.OP-AB.SW-0001.

Correct Answer: c

Proposed Change: Delete question based on no correct answer.

There is no correct answer to question #8. As shown on drawing 205342, Sheet 3, the 22SW39 is the 2B EDG Jacket Water and Lube Oil Coolers SW Supply Valve, which is normally **closed** with the EDG out of service. The 2A, 2B, and 2C EDGs are normally out of service, and no indication was given in stem that 2B EDG was in service. A leak on the EDG Lube Oil Cooler will not reduce SW Header pressure. Drawing 246689 shows that when the EDG is stopped the SW39 is demanded closed and does not go open until the EDG is started. There is an additional choice (b), which also had a leak location on a line which is normally out of service, so it cannot be assumed that the leak location was active.



with 22SW39-X (normal positi[on]) 2B EDG LO cooler leak would be invisible in control room.

SRO Q8

PRESSURE & TEMPERTURE RATINGS:
SEE ENGINEERING EVALUATION S-C-SW-MEE-1882 R0, FOR THE SERVICE WATER SIDE OF THE HEAT EXCHANGER. THE EVALUATION ESTABLISHES A NEW DESIGN PRESSURE AND TEMPERATURE AS LISTED IN THE TABLE BELOW

COMPONENT	COMPONENT TYPE	NEW DESIGN PRESSURE	NEW DESIGN TEMPERATURE
2CCE5	21 CC SHELL AND TUBE HTEX	LATER	LATER
2CVE39	22 CHARG PUMP GEAR OIL COOLER	200 PSI	300 DEG F
1CVE40	22 CHARG PUMP LUBE OIL COOLER	200 PSI	350 DEG F
2DAE1,2,3	NO. 2A, 2B 2C DIESEL GEN GENERATOR LUBE OIL COOLERS	200 PSI	150 DEG F
2DAE58,59,60	NO. 2A, 2B 2C DIESEL GEN JACKET WATER COOLERS	200 PSI	150 DEG F
2SJE181	22 SAFETY INJ. PMP LUBE OIL COOLER	200 PSI	350 DEG F
2SJV13	21 RHR PUMP ROOM COOLER	200 PSI	180 DEG F
2SVE14	22 CHARGING PUMP ROOM COOLER	200 PSI	180 DEG F
2SVE15	21 CONT. SPRAY PUMP ROOM COOLER	200 PSI	180 DEG F
2SVE16	21CC PUMP ROOM COOLER	200 PSI	180 DEG F
2SVE17	NO. 2 AUX. FW PUMP ROOM COOLER	200 PSI	180 DEG F

NOTE 'A' REV. 76 REVISED PER DCPs 80107664, AD M01/0 80107664, AD M12/0

REV. 75 80102290, ADs M40, M41, AND 80105796, AD M02/B.

205342-A-8763 SH. 5 (H-3)

205342-A-8763 SH. 4 (H-9)

GENERAL NOTES
USE PRINTS OF LATEST REVISION ONLY.
DO NOT SCALE - USE DIMENSIONS ONLY.
FOR LIST OF REFERENCE DRAWINGS SEE DRAWING NO. 205342-A-8763 SHEET 1
THIS DRAWING SUPERSEDES 205342-A-8763-20 242342-A-1966-1

(43)

FOR NOTES SEE SHEET 1 THIS DRAWING NO. 205342-A-8763.

ATTENTION: ANY REVISION TO THIS DRAWING SHALL BE MADE ONLY BY CAED

M600

SALEM NUCLEAR GENERATING STATION
NO. 2 UNIT
SERVICE WATER NUCLEAR AREA
P & ID MECH./CONT.

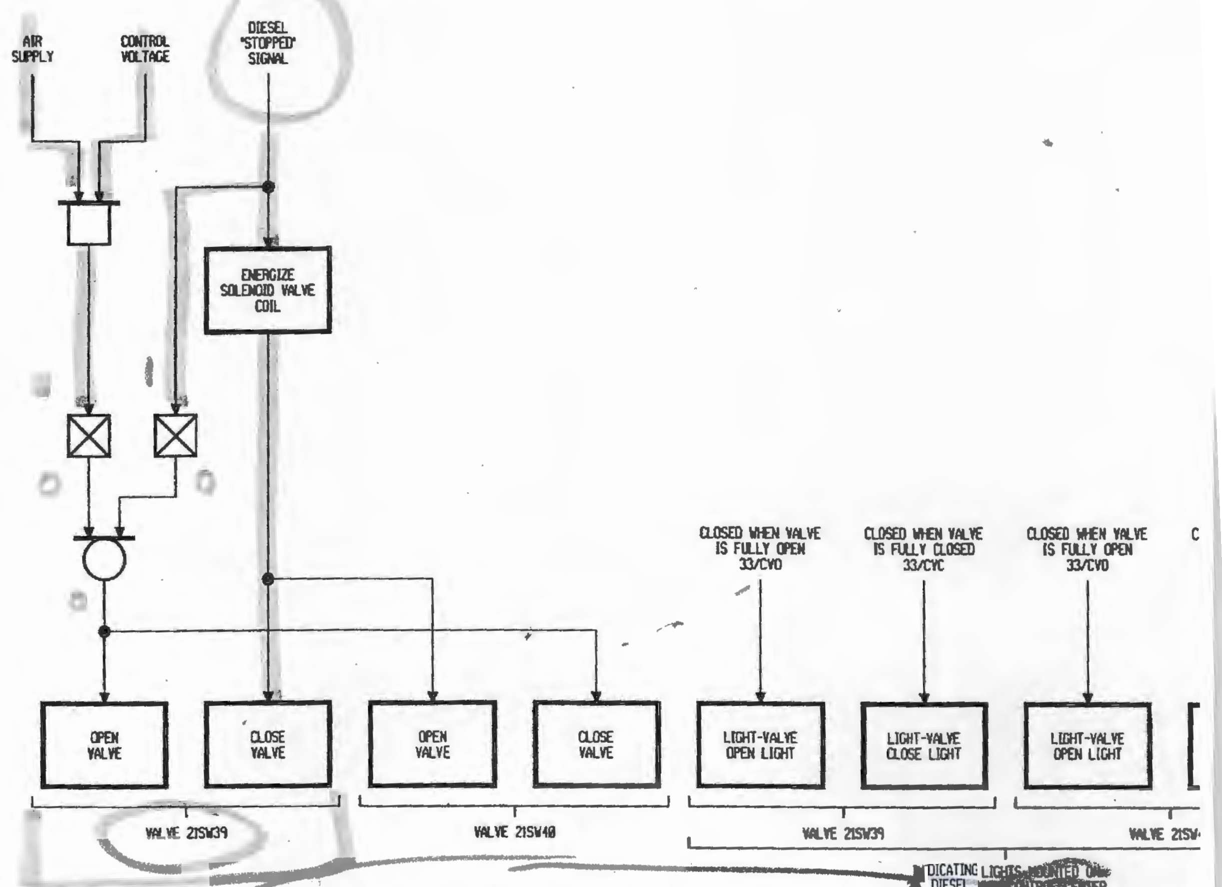
PSEG NUCLEAR, L.L.C.

DRAWN K. R. LEWIS CHECKED TF/ DN/V SCALE NONE
DATE 01-30-85 EXAMINED A. THOMSON
AUTH N1000 APPROVED R.T. STANLEY/MB

205342 A 8763-76
SH. 3

205342 A 8763

246689 B 9718 -03



NO. 2A DIESEL JACKET & LUBE OIL COOLERS SERVICE WATER SUPPLY [redacted] & DRAIN VALVE (VA. NO. 21SW40) SIMILAR FOR NO. 2B & 2C DIESELS

SEE TABLE I

TABLE I

DIESEL NO.	SUPPLY VA. NO.	DRAIN VA. NO.	SOLENIOD NO.	SCHEMATIC DWG. NO.	WIRING DIA DWG. NO
2A	21SW39	21SW40	SV-590	223696-BL-4042	226625-A-
2B	22SW39	22SW40	SV-591	223697-BL-4042	226626-A-
2C	23SW39	23SW40	SV-592	223698-BL-4042	226627-A-

REFERENCE DRAWINGS

SERVICE WATER NUCLEAR AREA PIPING DIAGRAM ____
NO. 22 HEADER NUCLEAR AREA INSTRUMENT SCHEMATIC SH
SERVICE WATER VALVES INDICATION SCHEMATIC ____
RECORDER PANEL 2RP1 WIRING DIAGRAM ________
NO. 2 UNIT SERVICE WATER SYSTEM MISC. VALVES EXTERN

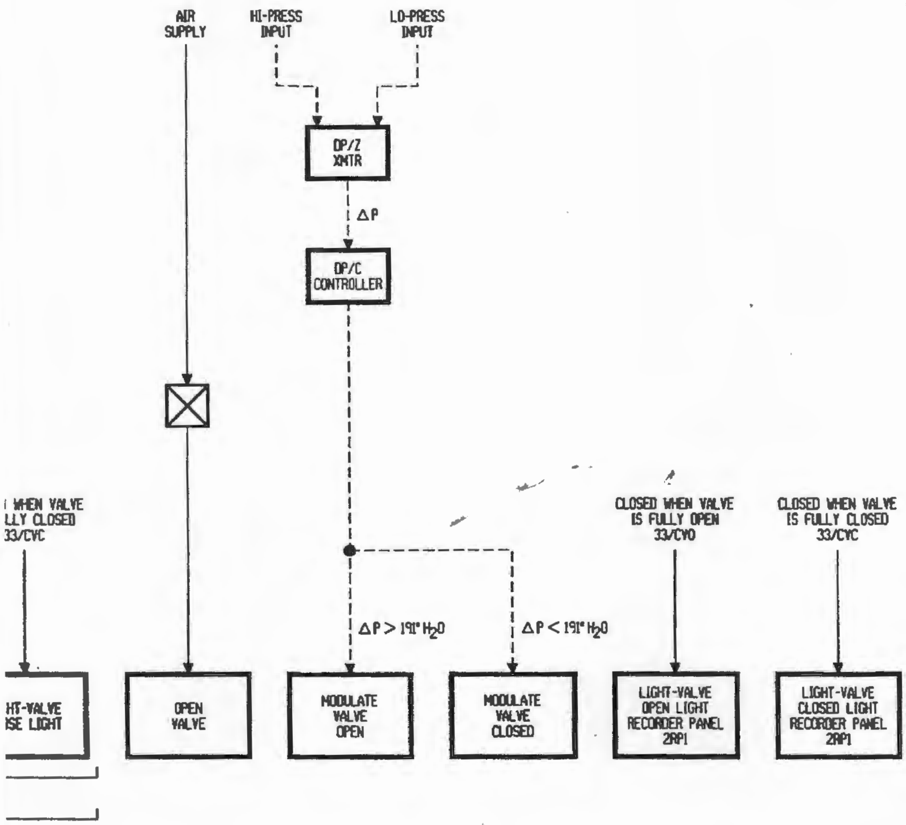
NOTE "A" - REV. 2
AUTH. N1000
REVISED & REDRAWN -
SUPERSEDES DWG. 246689-B-9718
REVISION 1
NO DESIGN CHANGE

NOTE "B" - REV. 3
REVISED PER
DCR 80054924, AD NO. 11/8

NO	DATE	DESCRIPTION	DWN	CKD	ENG	APP
3		SEE NOTE "B", ZONE A-2	GLK	TG	EG	RG
2		SEE NOTE "A", ZONE A-1	LGK	TF	AT	RTS

REVISION

5 6 7 8



SRO Q8

NO. 2A DIESEL JACKET & LUBE OIL COOLERS SERVICE WATER DISCHARGE VALVE (VA. NO. 21SW42) SIMILAR FOR NO. 2B & 2C DIESELS

DIGITAL LOGIC ———

ANALOG LOGIC – – –

NOTE:

FOR SYMBOL DESCRIPTION SEE DWG. NO. 224386-B-9567

DISCH. VA. NO.	DP/Z TRANSMITTER	DP/C CONTROLLER
21SW42	DP9628Z	DP9628C
22SW42	DP9630Z	DP9630C
23SW42	DP9632Z	DP9632C

—————— DWG. NO. 205342-A-8763
—————— DWG. NO. 240716-B-9658
—————— DWG. NO. 244142-B-9680
—————— DWG. NO. 244545-A-1637
D. ————— DWG. NO. 244991-A-1658

GENERAL NOTES
USE PRINTS OF LATEST REVISION ONLY.
DO NOT SCALE - USE DIMENSIONS ONLY.
FOR LIST OF REFERENCE DRAWINGS SEE
DRAWING NO THIS DRAWING
THIS DRAWING SUPERSEDES 246689-B-9718-1

ATTENTION: ANY REVISION TO THIS DRAWING SHALL BE MADE ONLY BY CAED — M600

SALEM NUCLEAR GENERATING STATION
NO. 2 UNIT-SERVICE WATER SYSTEM
DIESEL JACKET AND LUBE OIL COOLERS
SERVICE WATER VALVES

LOGIC DIAGRAM — CONTROLS

PSEG NUCLEAR, L.L.C.

DRAWN J.G. SMELTZER CHECKED T.E./L.T.S. SCALE NONE
DATE 6-9-86 EXAMINED A.T.
AUTH NU000 APPROVED R.T.S.

246689 B 9718 -03

246689 B 9718

5 6 7 8

Question RO#61

Given the following conditions:

- Unit 2 is operating at 100% power with an identified small fuel pin leak.
- A 5 gpm tube leak occurs on 22 SG

Of the following, which is the only radiation monitor that will NOT show a change from this tube leak?

a. 2R19B, 22 SG Blowdown
b. 2R46A-22 Main Steam Line
c. 2R15-condenser Air Ejector
d. 2R41D-Plant Vent Release Rate

Answer: b

Licensee Recommendation: The licensee proposed to delete this question based upon there is no correct answer. Although the 2R46 monitors were designed to provide continuous monitoring of high-level, post-accident releases of radioactive noble gases via the safety relief valves, the atmospheric dump valves, and the auxiliary feed pump turbine, they will exhibit a response to N-16 at levels corresponding to leak rates in the gallon per minute range according to DCP 80057587. Thus, these monitors can provide indication of a tube lead while the unit is at power.

NRC Conclusion: This question was initially acceptable for the examination based upon the (incorrect) premise that the 2R46 monitors would only provide indication during high levels of post-accident releases of radioactive noble gases. However, based upon documentation stating that these monitors can provide indication in response to N-16 during low steam generator tube leak rates, the NRC concluded that Choice b is incorrect. The NRC will delete this question based upon the determination that there is no correct answer.

Question SRO#6

Given the following conditions:

- Unit 1 is operating at 100% power.
- A breaker fault occurs on the 2-6 500 KV breaker.
- The 2-6 500 KV breaker does NOT trip, but should have.
- 15 seconds after the breaker failure, Unit 1 has not tripped.

Which of the following identifies how the Unit 1 CRS should proceed?

a. Direct the RO to manually trip the reactor and go to EOP-TRIP-1, Reactor Trip or Safety Injection. Concurrently with the EOP implementation, initiate S1.OP-AB.LOOP-0001, Loss of Offsite Power, and perform Attachment 2, Loss of Group Busses, Part A, Loss of 1E and 1H 4KV Group Buses.

b. Direct the RO to manually trip the reactor and go to EOP-TRIP-1. Concurrently with EOP implementation, initiate S1.OP-AB.LOOP-0001, and perform

Attachment 2, Loss of Group Busses, Part B, Loss of 1F and 1G 4KV Group Buses.

c. Enter S1.OP-AB.LOOP-0003, Partial Loss of Off-Site Power, then enter S1.OP-AB.CW-0001 Circulating Water System Malfunction, and perform a power reduction to 83% power or less.

d. Enter S1.OP-AB.LOOP-0003, then enter S1.OP-AB.CW-0001 and open the Hood Spray Bypass valves 11-13MC62s.

Answer: d

Licensee Recommendation: The licensee proposed that Choice c be accepted as an additional correct response. According to the licensee, several applicants assumed that for the given conditions that condenser delta-T would increase rapidly, therefore the Continuous Action Summary of S1.OP-AB.CW-0001 would come into effect because condenser delta-T would exceed 27 °F. This would require the crew to initiate a load reduction in accordance with S1.OP-AB.LOAD-0001, thus making Choice c to be a correct response. The applicants' assumption was substantiated by a simulator plant computer screen shot showing the rapid increase in condenser delta-T when the conditions provided in the question were run on the simulator and, within a short period of time, condenser delta-T reached 30°F.

NRC Conclusion: This question was initially acceptable for the examination based upon the (incorrect) premise that the Continuous Action Summary would not come into effect. However, based upon the simulator modeling of condenser delta-T for the given conditions, the NRC concluded that Choice c is a correct response. The NRC will accept Choices c and d as correct responses to this question.

Question SRO#8

Given the following conditions:

- Unit 2 is operating at 100% power.
- Operators receive the following alarms:
- OHA B-13 21 SW HDR PRESS LO
- OHA B-14 22 SW HDR PRESS LO
- SW header pressure indication in the control room reads 98 psig for both headers.

No other OHA's have annunciated.

Which of the following describes both the possible location of a Service Water System leak which would cause these indication, and how the CRS should respond?

Assume each of the leaks is large enough to cause the indications present in the control room.

a. 4 Service Water Bay. Split SW Bays by closing 21SW17 and 22SW17 IAW S2.OP-AB.SW-0003, Service Water Bay Leak.

b. Nuclear header x-over line between the 21SW23 and the 22SW23. Shut EITHER SW23 using Attachment 6, Service Water Valve Malfunctions, of S2.OP-AB.SW-0001.

c. 2B EDG Lube Oil Cooler. Isolate BOTH SW supply header isolation valves and BOTH return header isolation valves to 2B EDG IAW S2.OP-AB.SW-0001, Loss of Service Water Header Pressure.

d. 21 CCW HX end bell. Ensure 22 CC HX controller set lower than 21 CCW HX, and isolate 21 Service Water Header using Attachment 4.

Answer: c

Licensee Recommendation: The licensee proposed to delete this question based upon there is no correct answer. For the given plant conditions (no emergency diesel generators running), the service water supply header isolation valves (SW39s) would already be closed. These valves are normally closed and are open if their respective EDG is running. Therefore, if a leak developed on the 2B EDG Lube Oil Cooler (Choice c), it would not reduce SW header pressure because the oil cooler is already isolated. The proposal to delete this question is based up the SW flow path on Drawing 205342 Sheet 3 and the valve control logic on Drawing 246689.

NRC Conclusion: This question was initially acceptable for the examination based upon the (incorrect) premise that the service supply header isolation valves (SW39s) were in the open position. However, based upon the plant conditions provided in the question, and documentation supporting that these valves would have already been closed, The NRC concluded that Choice c is incorrect. The NRC will delete this question based upon the determination that there is no correct answer.