Draft Submittal

(Pink Paper)

SEQUOYAH NUCLEAR PLANT
EXAM 2002-301
50-327 & 50-328

DECEMBER 2 - 6, 2002

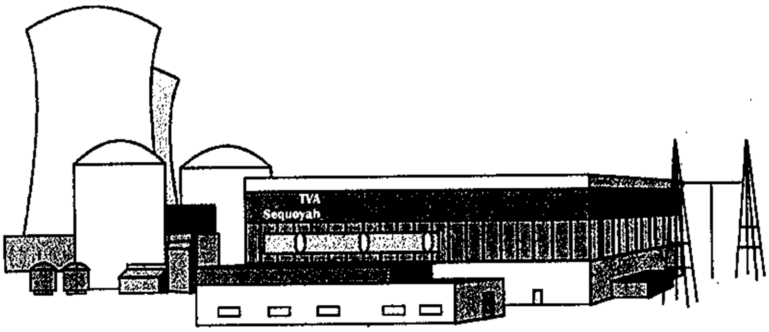
1. Reactor Operator Operator Written Exam

Draft Submittal
(Pink Paper)

SEQUOYAH NUCLEAR PLANT
EXAM 2002-301
50-327 & 50-328

DECEMBER 2 - 6, 2002

1. Senior Reactor Operator Written Exam



Sequoyah Nuclear Plant

12/2002 NRC EXAM

12/2002 NRC EXAM

Written Exam

- Cross-Reference
- RO Questions
- Cross-Reference
- SRO Questions
- RO Exam
- RO Key
- SRO Exam
- SRO Key

Tennessee Valley Authority TVA

RO/SRO/KA Cross Reference

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
001 AK1.03	AOP-C.01-B.1 002	B	C	A
001 K.301	001 K.301	B	A	N
001 K5.30	CONTROL*RODS 020	B	M	A
003 A1.03	RCP-B.12 002	B	M	A
004 A3.07	004 A3.07	B	C	N
005 A1.01	RHR-B.13.H 001	B	M	N
006 A1.07	ECCS-B.3 002	B	M	A
006 K6.19	006 K6.19	B	C	N
008 K4.02	CCS-B.9.A 001	B	A	A
009 EK1.01	INPO3 955	B	C	A
010 A2.02	010 A2.02	B	M	N
012 K6.07	012 K6.07	B	C	N
013 K4.01	013 K4.01	B	M	N
015 AK2.08	AOP-R.05-B.5 001	B	M	A
016 A3.02	RCS-TEMP-B.2 008	B	M	A
016 K1.01	INCORE-B.1.B 002	B	M	A
017 K6.01	INCORE-B.1.D 003	B	M	A
022 K2.01	022 K2.01	B	C	N
024 AA1.26	ES-0.1-B.1 002	B	C	A
025 AK2.05	FR-Z.1-B.2 001	B	C	A
025 K5.02	CTMT-B.11 001	B	C	A
027 AK1.02	OPL271C353.4 001	B	C	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
029 EK2.06	RPS-B.5.B 001	B	C	A
034 A1.02	034 A1.02	B	A	N
034 K4.02	FH-B.12.B 001	B	M	A
035 K5.01	035 K5.01	B	C	N
037 2.1.7	AOP-R.01-B.2 003	B	A	A
037 AA1.11	AOP-R.01-B.2 004	B	M	A
038 EK3.08	INPO2 976	B	C	A
039 K3.04	FW-B.5 001	B	A	A
040 AK1.01	INPO2 231	B	C	A
051 AK3.01	SDCS-B.12 023	B	C	A
058 AK3.01	AOP-P.02-B.4 002	B	C	A
059 A4.01	059 A4.01	B	C	N
059 K1.04	MFW 001	B	M	A
060 AK2.02	060 AK2.02	B	C	N
061 AK2.01	061 AK2.01	B	A	N
061 K2.01	061 K2.01	B	M	N
061 K6.01	061 K6.01	B	M	N
062 A4.03	INPO8 155	B	M	A
063 A2.01	063 A2.01	B	M	N
063 K4.04	FW-B.5.B 004	B	M	A
064 K2.02	064 K2.02	B	C	N
068 A3.02	RADWASTE-B.12 008	B	M	A
069 AA1.03	CTMT-B.5 004	B	C	A
071 A1.06	RMS-B.9 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
072 A1.01	RMS-B.2 004	B	M	A
074 EK1.01	FR-C.1-B.2 003	B	C	A
074 EK2.02	FR-C.1-B.2 011	B	A	A
075 K1.01	RCW-B.9 001	B	M	A
076 AK3.05	076 AK3.05	B	C	N
078 K1.01	AIR-B.12 002	B	C	A
079 K4.01	AIR-B.5 014	B	C	A
103 A3.01	103 A3.01	B	M	N
2.1.24	2.1.24	B	A	N
2.2.11	CTMT-B.11 004	B	M	A
2.2.4	2.2.4	B	A	N
2.3.11	ODCM-B.5 001	B	C	A
2.3.9	CTMT PURGE-B.4 001	B	M	A
2.4.39	REP-B.1.D 002	B	M	A
2.4.4	AOP-R.02-B.2 001	B	C	A
E01 EA1.1	ES-0.0-B.3 001	B	C	A
E03 EA1.2	ES-1.2-B.2 006	B	C	A
E03 EK1.3	ES-1.2-B.2 004	B	M	A
E04 EK2.2	ECA-1.2-B.1 002	B	M	A
E05 EA1.3	FR-H.1-B.3 004	B	C	A
E08 EK3.2	FR-P.1 001	B	C	A
E09 EA1.1	OPL271C382.4 001	B	A	A
E10 EK3.1	ES-0.2-B.3 003	B	C	A
E14 EK1.3	ECCS-B.2 002	B	M	A
E15 EK1.1	FR-Z.2-B.2 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
003 K4.02	T/S0304.02 003	R	C	A
005 AA2.01	005 AA2.01	R	C	N
008 A2.04	008 A2.04	R	C	N
012 K2.01	RDCNT-B.7.B 001	R	C	A
015 A2.03	FISSION*PROD*POISON 052	R	M	A
022 2.4.27	022 2.4.27	R	A	N
025 2.2.13	025 2.2.13	R	M	N
029 A1.02	PI-B.10 001	R	M	A
039 K5.05	RVINT-B.8 001	R	M	A
056 A2.04	056 A2.04	R	C	N
056 AK3.01	OPL271C368.3 001	R	M	A
056 K1.03	056 K1.03	R	M	N
059 AA2.05	059 AA2.05	R	M	N
064 A3.06	D/G-B.10 002	R	A	A
067 AA2.17	AOP-C.04 001	R	M	A
068 AK2.07	D/G-B.6 012	R	M	A
068 K4.01	RADWASTE-B.12 006	R	M	A
071 A4.26	WGDS-B.12 001	R	C	A
078 K3.03	078 K3.03	R	C	N
079 2.1.1	079 2.1.1	R	M	N
086 K3.01	086 K3.01	R	C	N
2.1.16	2.1.16	R	M	N
2.1.31	PZR PRESS-B.9 006	R	C	A
2.2.26	REFUELING-B.1.G 001	R	M	A
2.3.10	2.3.10	R	C	N

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.4.20	AOP-M.01-B.3 002	R	M	A
2.4.5	EPM-4-B.7 001	R	C	A
E01 2.2.25	E01 2.2.25	R	C	N
E15 EK2.2	E15 EK2.2	R	C	N

003 DROP ROD 2.4.4	AOP-C.01-B.5 011	S	M	A
004 2.4.1	OPL271C367.1 002	S	C	A
007 2.1.10	PRT-B.7 004	S	M	A
010 2.4.47	AOP-I.04-B.2 001	S	A	A
011 2.1.6	OPL271C367.1 003	S	C	A
013 2.4.47	E-0-B.6 003	S	M	A
015 AA2.07	AOP-R.04-B.5 001	S	A	A
022 AA2.01	AOP-R.05-B.2 001	S	A	A
025 AA2.06	RHR-B.12.A 003	S	M	A
027 AA2.04	T.S-2.1-B.2 001	S	A	A
040 AA2.01	OPL271C379.3 001	S	A	A
056 AA2.18	E-0-B.3.A 007	S	C	A
069 2.1.14	069 2.1.14	S	C	N
2.1.10	OPL271C458.1 001	S	M	A
2.1.22	GO-2-B.1 003	S	M	A
2.1.7	PZR LEVEL-B.14 003	S	A	A
2.2.25	OPL271C180.3 001	S	M	A
2.2.9	SPP-9.5 001	S	M	A
2.3.1	RCI-15 001	S	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.3.2	RADIATION 002	S	M	A
2.3.3	AOP-C.04-B.5 008	S	M	A
2.4.33	OPDP-4 002	S	M	A
2.4.45	AOP-M.03-B.1 002	S	C	A
E01 EA2.1	ES-0.0-B.5 002	S	C	A
E04 EA2.1	ECA-1.2-B.2 001	S	C	A
E09 EA2.1	ES-0.2-B.3 001	S	C	A
E12 2.4.16	ECA-2.1-B.1 004	S	A	A
E15 EA2.1	E15 EA2.1	S	M	N
E16 2.4.41	E16 2.4.41	S	C	N

1. 001 K3.01 001

The unit is at 100% power with no Xenon transients and all systems functioning normally. When the following annunciators are received:

"TS-68-2P/Q Reac Cool Loops T Ref T Auct High-Low"
"Rod Control System Urgent Failure"

Tref is 4°F above Tave.

Which one of the following is correct for this condition?

A. Raise turbine load and borate.

B. Lower turbine load and borate.

C. Raise turbine load or borate.

✓D. Lower turbine load or dilute.

A. Incorrect, per AOP-C.01 (Tref is > Tave). Step 1 RNO only allows load change OR change in RCS boron concentration. Increasing turbine load is not correct (Tref is > Tave). Boration is not correct (Tref is > Tave).
B. Incorrect, Step 1 RNO only allows load change OR change in RCS boron concentration. Boration is not correct (Tref is > Tave).
C. Incorrect, per AOP-C.01 (Tref is > Tave).
D. Correct, per AOP-C.01 (Tref is > Tave). Step 1 RNO only allows load change OR change in RCS boron concentration. Either lower turbine load (Tref is > Tave) OR dilute (Tref is > Tave).

K/A[CFR]: 001 K3.01 [2.9/3.0] [41.7]

Reference: 1-AR-M5-A (C-6)
AOP-C.01 R8 section 2.1 step 2 RNO.

LP/Objective: OPL271RDCNT B.9

History: New question.

Level: Analysis

Comments: FHW 12/02 001 K3.01

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Failure of a Control Bank to Move in AUTO

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**ENSURE** rod control in MAN.	
2.	**POSITION** control rods to minimize T-avg. - T-Ref deviation.	**PERFORM** the following: • **ADJUST** turbine load **OR** • **ADJUST** RCS boron concentration.
3.	**MONITOR** following parameters to ensure core power distribution is within normal limits: • Power range NIS • T-avg. and ΔT channels • Incore T/Cs • Delta flux (ΔI)	

Source	Setpoint	TS-68-2P/Q REAC COOL LOOPS T REF T AUCT HIGH-LOW
SER 2114, 2115 TS-68-2P/Q, TS-68-2Q/P	Tref ± 3°F compared with Tauct.	

Probable Causes

1. Normal heatup, cooldown, power ascension or descension inprogress with controls in MANUAL.
2. The reactor control system, steam dump system or S/G PORV's are not maintaining Tavg on program.
3. Inadvertent dilution or boration.
4. Channel testing and/or malfunctioning.
5. Secondary system leak.

Corrective Actions

[1] **COMPARE** plant indicators to verify validity of alarm.

[2] **IF** controls are in AUTO when alarm occurs, **THEN**

PLACE rod control system (1-HS-85-5110) in manual and match Tavg with Tref.

[3] **IF** rod control system is malfunctioning, **THEN**

GO TO AOP-C.01, Rod Control System Malfunctions.

[4] **IF** a boron concentration change is suspected, **THEN**

GO TO AOP-C.O2, Uncontrolled RCS Boron Concentration Changes.

[5] **IF** a steam line or feedwater line break or leak is suspected, **THEN**

GO TO AOP-S.05, *Steam Line or Feedwater Line Break/Leak.*

[6] **IF** Tavg channel failed, **THEN**

GO TO AOP-I.02, RCS Loop RTD Instrument Malfunction.

[7] **IF** Tref (PI-1-73) failed, **THEN**

GO TO AOP-I.08, Turbine Impulse Pressure Instrument Malfunction.

[8] **IF** alarm is valid, Reactor is critical, and Tavg < 551 °F, **THEN**

PERFORM 0-SI-SXX-068-127.0, *RCS and Pressurizer Temperature and Pressure Limits.*

[9] **EVALUATE** Technical Specifications (3.3.1 and 3.3.2).

References

45B655-05A-0,
47B601-68-4

INTRODUCTION:

The chemical and volume control system is a major auxiliary system that functions to control RCS inventory, to maintain RCS chemistry, to provide the reactor coolant pumps with seal water, and to control the soluble poison concentration of the RCS. In addition, the CVCS supplies borated coolant to the RCS in the event of an accident.

Major system interfaces include the pressurizer level control system, which varies charging flow based upon pressurizer level. The reactor make-up system that allows boric acid, demineralized water, and corrosion inhibiting chemicals to be added via the CVCS during normal at-power and shutdown conditions. The emergency core cooling systems uses a portion of the system piping and the charging pumps to provide high-pressure injection of borated water during accident conditions.

Purposes

1. Maintains programmed pressurizer level;
2. Controls reactor coolant chemistry and activity;
3. Adjusts and controls reactor coolant boron concentration;
4. Provides reactor coolant make-up;
5. Supplies seal water to the reactor coolant pumps;
6. Portions used for emergency core cooling and emergency boration;
7. Used to fill and hydrostatically test the reactor coolant system.

Subsystems

1. Charging, letdown, and seal water system
2. Chemical control, purification, and make-up system

Design Bases

1. Reactivity control
2. Boric acid used as a neutron absorber for following reactivity changes:
3. RCS temperature change between cold shutdown and hot standby.
4. Burnup of fuel and burnable poisons.
5. Xenon transients.
6. Buildup of fission products in the fuel.

Make-up control

1. System capable of boration of RCS from either of two paths and from either of two boric acid sources.
2. Amount of boric acid stored exceeds amount needed to :
3. Borate RCS to cold shutdown with highest worth rod stuck out.
4. Go to hot shutdown.
5. Compensate for subsequent xenon decay.
6. System capable of counteracting inadvertent positive reactivity due to maximum boron dilution accident.

RCS inventory regulation

Pressurizer inventory maintained in allowable range for all operating modes as well as small RCS leaks.

The charging and letdown capacities permits heat-up and cool-down of RCS at designed rate with pressurizer level normal.

4. 006 K6.19 001

This question has reference material attached.

Unit One is operating at 100% power. On January 10 at 0200 hours the 1 A-A CCP failed to start on an attempted pump swap. The crew entered the appropriate LCO action. Maintenance can **NOT** get replacement parts until January 14.

Which one of the following is the correct Tech Spec action?

✓A. Be in Hot Standby on January 13 at 0800 hours.

B. Be in Hot Standby on January 13 at 1400 hours.

C. Be in Hot Shutdown on January 14 at 0800 hours.

D. Be in Hot Shutdown on January 14 at 0200 hours.

A. Correct, the action is 72 hrs to operability or Hot Standby in the next 6 hours. January 10 0200 hrs. to January 13 0200 is 72 hours plus 6 hours is January 13 at 0800.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]:	006 K6.19 [3.7/3.9] [41.7]
Reference:	Tech Specs 3.5.2.a.
LP/Objective:	OPL271C079 B.2
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 006 K6.19 **Provide a copy of Tech Specs 3.5.2**

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} Greater Than or Equal to 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

SURVEILLANCE REQUIREMENTS (Continued)

Valve Number	Valve Function	Valve Position
a. FCV-63-1	RHR Suction from RWST	open
b. FCV-63-22	SIS Discharge to Common Piping	open

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Deleted. R143

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal and automatic switchover to containment sump test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

   a) Centrifugal charging pump

   b) Safety injection pump

   c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1.	Centrifugal charging pump	Greater than or equal to 2400 psig
2.	Safety Injection pump	Greater than or equal to 1407 psig
3.	Residual heat removal pump	Greater than or equal to 165 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months. R144

Charging Pump Injection Throttle Valves	Safety Injection Cold Leg Throttle Valves	Safety Injection Hot Leg Throttle Valves
Valve Number	Valve Number	Valve Number
1. 63 - 582	1. 63 - 550	1. 63-542
2. 63 - 583	2. 63 - 552	2. 63-544
3. 63 - 584	3. 63 - 554	3. 63-546
4. 63 - 585	4. 63 - 556	4. 63-548

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

1. For safety injection pump lines with a single pump running:

   a. The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 443 gpm, and R144

   b. The total pump flow rate is less than or equal to 675 gpm.

2. For centrifugal charging pump lines with a single pump running:

   a. The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 309 gpm, and R144

   b. The total pump flow rate is less than or equal to 555 gpm.

3. For all four cold leg injection lines with a single RHR pump running a flow rate greater than or equal to 3931 gpm. R144

6. 010 A2.02 001

The unit is operating at 100% power when the controlling pressurizer pressure channel fails high.

Which one of the following is the correct system response and the required action to mitigate the event?

A. Only the pressurizer spray valve associated with the failed channel will open and the operators manually closes this valve.

✓B. Both pressurizer spray valves will open and the operator manually closes these two valves.

C. No pressurizer spray valves will open due to a pressurizer pressure channel bistable interlock and the operator places the spray valve controllers in manual.

D. Both pressurizer spray valves will open and the auxiliary pressurizer ~~spray~~ valve will open if selected to the failed channel and the operator manually closes all three valves.

Δ to PORV open

A. Incorrect the pressurizer pressure controller will open both spray valves.
B. Correct per references.
C. Incorrect the bistable is associated with the PORV.
D. Incorrect the pressurizer pressure control channels do not have input to the auxiliary spray valve.

K/A[CFR]:	010 A2.02 [3.9/3.9] [41.5 43.5]
Reference:	1-47W611-68-3 AOP-I.04 section 2.1.
LP/Objective:	OPL271PZRPCS B.14
History:	New question.
Level:	Memory
Comments:	FHW 12/02 010 A2.02

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Pressurizer Pressure Instrument Malfunction

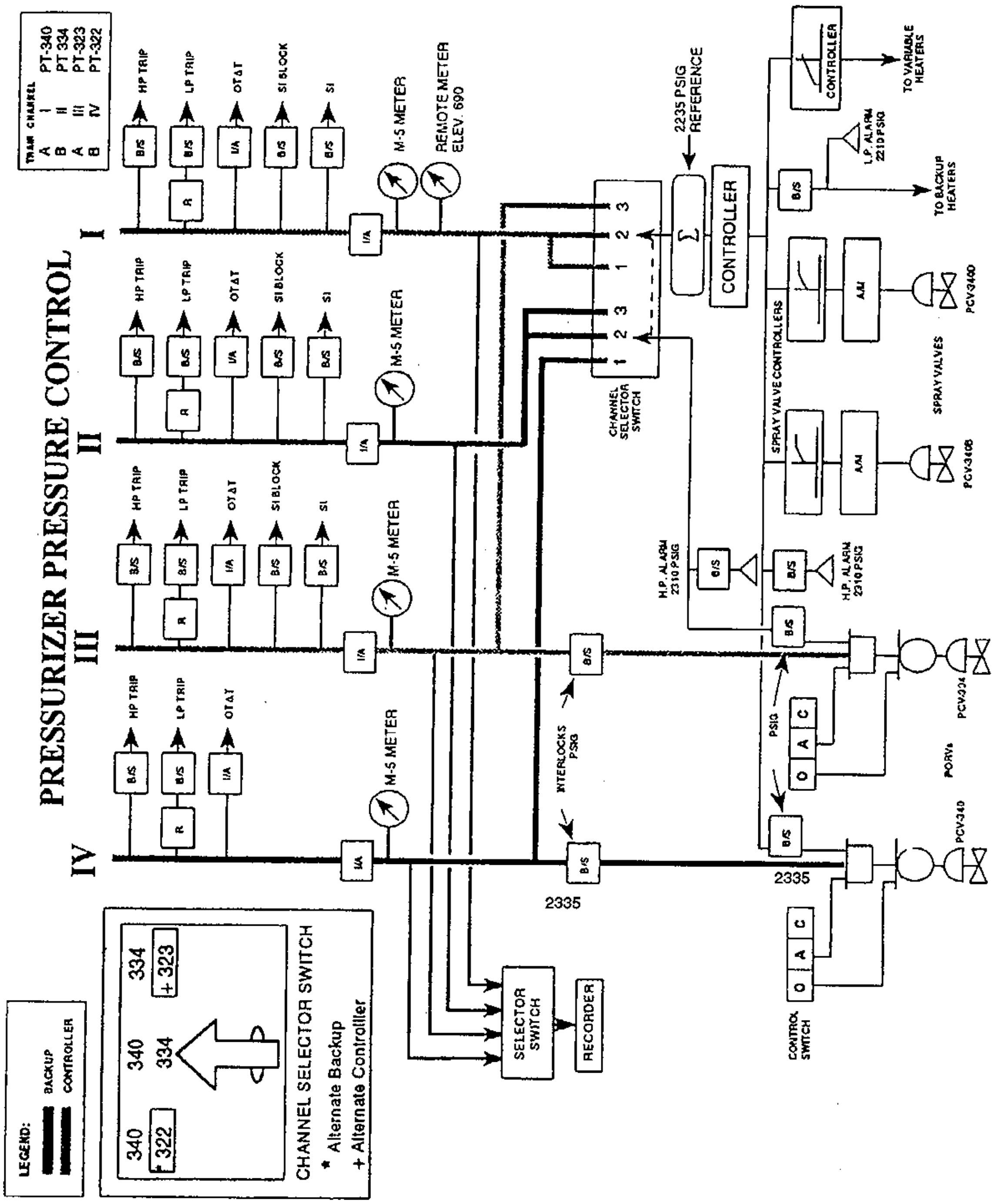
NOTES:

- A failure of channel III (P-68-323) will affect the automatic actuation of PCV 68-334, PZR PORV, in the normal pressure control circuit. LTOPS operation of this PORV is unaffected by this failure.
- A failure of channel IV (P-68-322) will affect the automatic actuation of PCV 68-340A, PZR PORV, in the normal pressure control circuit. LTOPS operation of this PORV is unaffected by this failure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**MONITOR** pressurizer pressure stable or trending to desired pressure.	**RESTORE** pressurizer pressure **USING** manual control of the following: • PZR Spray controllers PIC-68-340D (Loop 1) **AND/OR** PIC-68-340B (Loop 2) **OR** PIC-68-340A **OR** • Pressurizer Heaters

APPENDIX H

PRESSURIZER PRESSURE CONTROL



7. 012 K6.07 001

Unit One is operating at 100% power. Instrument Mechanics are performing a surveillance which required removing RCS Loop 1 ΔT/Tavg instrument T-68-2 from service. The bistables associated with this loop have been placed in the tripped position.

Which one of the following is correct for this condition?

A. If loop 2 ΔT/Tavg instrument T-68-25 was removed from service and the bistables associated with this loop were tripped; Unit One would remain at power after a OTΔT turbine runback.

✓B. If loop 2 ΔT/Tavg instrument T-68-25 was removed from service and the bistables associated with this loop were tripped; Unit One would trip.

C. Unit One will remain at power since the runback/trip logic has now changed to 2 of 3 remaining loops.

D. Tech specs will require Unit One to be in Hot Standby if the RCS Loop 1 ΔT/Tavg instrument T-68-2 is **NOT** declared operable within 1 hour.

A. Incorrect, if all bistables associated with a second loop were tripped then the 1 of 3 trip logic would be made and the unit will trip.
B. Correct, if all bistables associated with a second loop were tripped then the 1 of 3 trip logic would be made and the unit will trip.
C. Incorrect, the runback/trip logic is changed to 1 of 3 loops.
D. Incorrect, per TS 3.1.1.1 Table 3.3-1. 7., action 6 the required action time is 6 hours.

K/A[CFR]:	012 K6.07 [2.9/3.2] [41.7]
Reference:	AOP-I.02 R1 Appendix A page 5 of 12. Tech Specs 3.3.1.1 Table 3.3-1 and action 6.
LP/Objective:	OPL271RPS B.19
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 012 K6.07

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE. R194

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel and interlock shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1. R16

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceeding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3.1. R255

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	1
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature Delta T Four Loop Operation	4	2	3	1, 2	6#
8. Overpower Delta T Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	6#

R145

TABLE 3.3-1 (Continued)

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

d. Above 10% of RATED THERMAL POWER, the provisions of Specification 3.0.3 are not applicable.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, operation may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. R51

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.1. R51

ACTION 7 - Deleted.

R145

APPENDIX A

REMOVING RCS LOOP 1 ΔT/TAVG INSTRUMENT LOOP T-68-2 (T-411/412) FROM SERVICE

A. Setup (cont'd)

5. (Continued)

ANNUNCIATORS [XA-55-4]	CHECK (√)	
	DARK	LIT
PROTECTION SET II BYPASS [B-7]		
PROTECTION SET III BYPASS [C-7]		
PROTECTION SET IV BYPASS [D-7]		

6. **VERIFY** Steps 1 through 5 COMPLETE. ______________

CAUTION: **If any loop 2, 3, or 4 ΔT/T-avg trip status light in Step 5 is LIT, completion of this Appendix will initiate one or more of the following signals:**

- **OPΔT reactor trip**
- **OTΔT reactor trip**
- **OPΔT turbine runback/rod withdrawal block**
- **OTΔT turbine runback/rod withdrawal block**
- **Feedwater isolation on low T-avg with reactor trip**
- **Loss of P-12 permissive - unblock steam dumps**

7. **NOTIFY** Unit Operator to VERIFY current plant status allows removing this ΔT/TAVG channel from service.

____
UO Initials

8. 013 K4.01 001

Which one of the following is correct concerning ~~the reset of~~ a Safety Injection signal?

A. Low Pressurizer pressure input to Safety Injection logic can be manually blocked at any time.

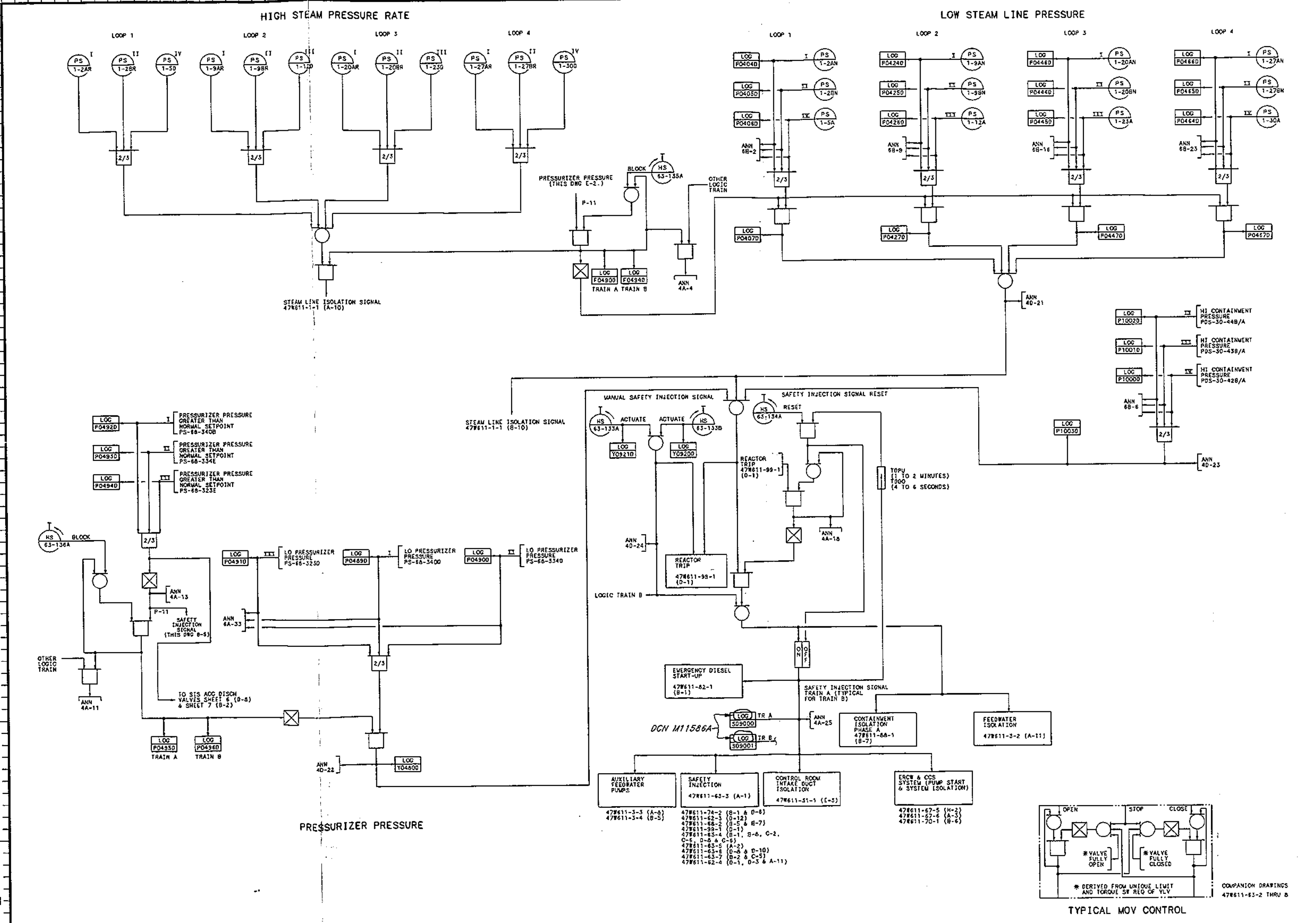
B. Hi Containment pressure input to Safety Injection logic can be manually blocked at any time.

C. Safety Injection can be reset immediately after automatic actuation.

✓D. If the Safety Injection has been reset and the reactor is tripped, all subsequent automatic Safety Injection actuation signals are blocked.

A. Incorrect per reference, manually blocked below P-11.
B. Incorrect per reference, no manual block.
C. Incorrect per reference, time delay pick up before reset logic can be made.
D. Correct per reference.

K/A[CFR]:	013 K4.01 [3.9/4.3] [41.7]
Reference:	47W611-63-1
LP/Objective:	OPL271RPS B.16
History:	New question.
Level:	Memory
Comments:	FHW 12/02 013 K4.01



NOTES:

1. FOR SYMBOLS OTHER THAN THOSE NOTED BELOW SEE INSTRUMENTATION AND IDENTIFICATION STANDARDS, LATEST ISSUE.
2. FOR COMPLETE INSTRUMENTATION AND COMPONENT SEPARATION DESIGNATIONS SEE CONTROL DIAGRAM 47W610-63-1 & 2.
3.
4. P1 = PRESSURE IN LOOP 1
   P2 = PRESSURE IN LOOP 2
   P3 = PRESSURE IN LOOP 3
   P4 = PRESSURE IN LOOP 4
5. PROTECTION SET CHANNEL DESIGNATIONS ARE ROMAN NUMERALS I, II, III & IV.
6. ANNUNCIATOR POINTS ON LOGIC DIAGRAMS ARE INDICATED BY GIVING THE ANNUNCIATOR PANEL NUMBER AND THE WINDOW NUMBER AS GIVEN ON TVA DRAWINGS 47B601-55-0 THRU 65.
7. DIGITAL AND ANALOG LOGIC SYMBOLS ARE USED ON LOGIC DIAGRAMS TO FUNCTIONALLY DESCRIBE THE PROCESS CONTROL REFER TO THE ASSOICATED WIRING SCHEMATIC FOR THE ELECTRICAL COMPONENTS USED TO IMPLEMENT THE CONTROL SCHEME.

NOTES CONTINUED ON SHEETS 2 THRU 8

SYMBOLS:

✱- LOCATED ON ELECTRICAL SWITCHGEAR.
#- LOCATED IN AUXILIARY CONTROL ROOM.
▲- TEST SWITCH.
△- LOCAL CONTROL SWITCH.

REFERENCE DRAWINGS:

LOGIC INDEX
47W611-0-1 & 2

MECHANICAL LOGIC DIAGRAMS
47W611-62-4
47W611-63-3 THRU 7
47W611-67-1,5 & 7
47W611-68-3
47W611-72-1
47W611-74-2
47W611-78-1
47W611-99-1

MECHANICAL CONTROL DIAGRAMS
47W610-1-1,2
47W610-30-1
47W610-63-1 & 2
47W610-68-1 THRU 4

NON-INTENT

WESTINGHOUSE FUNCTIONAL DIAGRAMS
5655028 SERIES
1082H70 SERIES

FOR ORIGINAL SIGNATURES IN TITLE BLOCK SEE REVISION 0 MICROFILM COPY.

REV	ECN/DCN	CHG DOC	DRWN	CHKD	APPD	DATE
4	DCN M11586A NON-INTENT	N/A N/A	DCR			

INC: DCA-M11586-5168 R0
INC: NON-INTENT CHANGE TO REMOVE 1-TAB REFERENCE DRAWING.

SCALE: NONE — EXCEPT AS NOTED

POWERHOUSE
UNIT 1 — CATEGORY 1

MECHANICAL LOGIC DIAGRAM SAFETY INJECTION SYSTEM

SEQUOYAH NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
NUCLEAR ENGINEERING
Q

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER: JOHN AVITT	CHECKER: E.MATTHEWS	R0 ISSUE PER: CCD REC SHEET NO. 7094. DRAWING MADE A CCD FROM 1,2-47W611-63-1 R5.	1 W.L.SUTTON
DESIGNER: N/A	REVIEWER: M.F.SCRUGGS		2 L.W.A./M.R.S.
			3 E.TURNER

ISSUED BY: P.G.TRUDEL

DATE 5-19-89 | 45 | M | CCD NO: 1-47W611-63-1 R4

PROCAD MAINTAINED DRAWING
THIS CONFIGURATION CONTROL DRAWING IS MAINTAINED BY THE SQN CAD UNIT AND IS NOW PART OF THE TVA PROCADAM DATABASE.

CONTROL ROOM DWG

10. 022 K2.01 001

Plant Conditions:
- The unit is at 100% power.
- 6.9 kv Shutdown Board "B" trips on a differential fault.
- All D/Gs functioned as designed.

Which one of the following is a result of this condition?

A. Feedwater flow is increased due to more demand by the S/G level program.

B. Train "B" RHR pump will be available when the blackout sequencer times out.

✓C. Available forced flow from the upper compartment coolers is reduced.

Runback will occur due to Loss of Heater Drain Tank pump.

D. Unit ~~load is procedurally reduced to 80% power.~~

A. Incorrect, AOP-P.05/06 requires manual reactor trip, therefore feedwater flow is not increased. S/G level is vital power and not load shed.
B. Incorrect, loads are not available from a board that had a differential fault. BOX and BOY relays will not pick up due to dead bus.
C. Correct upper compartment fans that are fed from the associated reactor vent board are no longer available.
D. Incorrect, AOP-P.05/06 power reduction guidance not given.

K/A {CFR}:	022K2.01 [3.0/3.1] [41.7]
References:	1,2-15E500-1 1,2-45N724-1 1,2-45N755-3 AOP-P.05/06 App T.
LP/Objectives:	OPN218E.012 B.4
History:	New Question
Level:	Comprehension
Comments:	FHW 12/02 022K2.01

APPENDIX T

480V REACTOR BUILDING VENT BOARD 1B-B LOAD LIST

COMPT	COMPONENT / LOAD
1C	RCP 2 Motor Heater
1D1	Incore Flux Drive Unit 1A
1D2	Incore Flux Drive Unit 1B
1E	Incore Flux Drive Unit 1C
2A	Containment Purge Air Exhaust Fan 1B
2C	RCDT Pump 1B (Note 1)
3B	H2 Recombiner 1B-B (Note 1)
3C	Incore Instrument Room Purge Supply Fan
3D	Incore Instrument Room Purge Exhaust Fan
3E	RCP 2 Oil Lift Pump
4A	Containment Instrument Room Unit Heater 1B
4B	Containment Floor and Equipment Drain Sump Pump 1B (Note 1)
4C	Ice Condenser Floor Cooling Pump 1B (Note 1)
4D	Ice Condenser AHUs (Note 1)
5A	Reactor Upper Compartment Unit Heater 1B
5E	RCP 4 Oil Lift Pump
6A	Ice Condenser Bridge Crane
6B	Spreading Room Exhaust Fan B
6C	RCP 4 Motor Heater
6D	Reactor Lower Compartment Unit Heater 1B
6E	Ice Condenser AHUs (Note 1)
7C	Containment Purge Air Supply Fan 1B
7E1	Control Bay Sump Pump 1 (Note 1)
7E2	Reactor Building Jib Crane

Note 1: If required, start redundant train equipment.
Note 2: Monitor temperatures locally.

APPENDIX T

480V REACTOR BUILDING VENT BOARD 1B-B LOAD LIST

COMPT	COMPONENT / LOAD
8A	Reactor Upper Compartment Cooler Fan 1B (Note 1)
8C	Reactor Upper Compartment Cooler Fan 1D (Note 1)
8E1	Water Intake Structure Heat Trace Cabinet (Note 2)
8E2	Reactor Upper Compartment Unit Heater 1D
9A	ERCW Cooling Tower Makeup Pump
9C	RCC Change Hoist
9E1	Equipment Hatch Hoist
10B	Ice Condenser End Wall Door 1B
10C	Ice Condenser Floor Defrost Heater 1B

12. 034 A1.02 001

This question has reference material attached.

Unit ~~One~~ is in mode 6 and currently performing 0-GO-13. The off going shift advised you the upper and lower internals storage area was dry and is in process of filling the refuel cavity.

The Reactor Cavity AUO said the level is on ladder rung number 31.

How many gallons of water is required to reach ladder rung number 21?

A. Approximately 19,000 gallons.

B. Approximately 67,000 gallons.

✓C. Approximately 99,000 gallons.

D. Approximately 143,000 gallons.

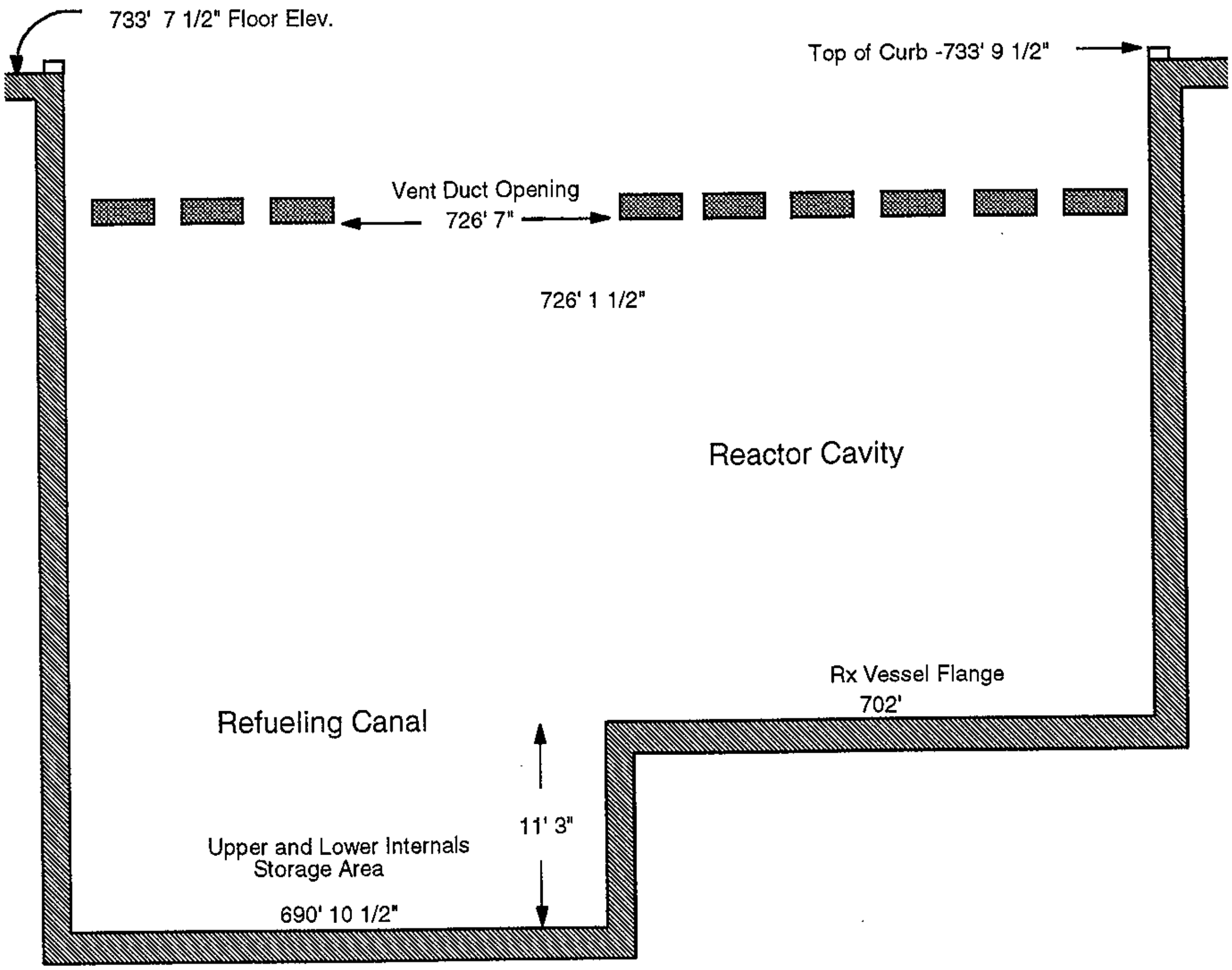
A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference, per appendix K page 2 each ladder rung is one foot.
Per Appendix K page 1 each foot above elevation 702 requires 9,889 gallons.
Per Appendix K page 2 ladder rung 31 is elevation 702. Rung 31 - 21 is 10 rungs.
10 rungs (10 feet) times 9,889 gallons is approximately 99,000 gallons.
D. Incorrect per reference.

K/A[CFR]:	034 A1.02 [2.9/3.7] [41.5]
Reference:	0-GO-13 R38 Appendix K pages 1 and 2.
LP/Objective:	OPL271C057 B.5
History:	New question.
Level:	Analysis
Comments:	FHW 12/02 034 A1.02 **Provide a copy of 0-GO-13 R38 Appendix K pgs 1 and 2**

Need to ADDRESS ~~stat~~ on FHS to meet K/A

APPENDIX K
Page 1 of 2

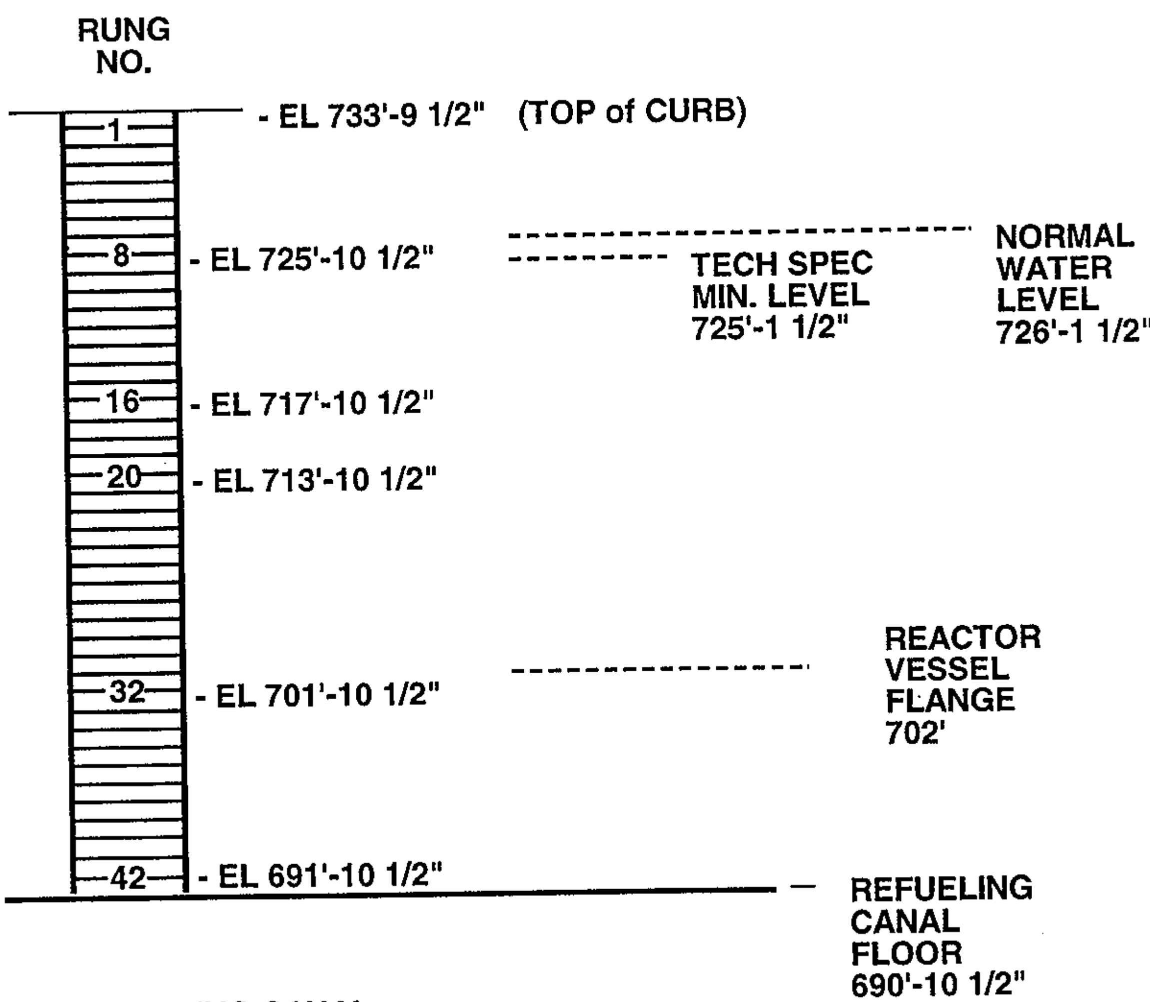
REACTOR CAVITY/REFUELING CANAL DIAGRAM



Total gallons required to fill Upper and Lower Internals Storage Area from el. 690' 10 1/2" to Rx. Vessel Flange el. 702'.	56,782 gals.
Total gallons required to fill Rx Cavity from ele 702' to 703' (1 ft ele = 9889 gals when ≥ ele 702')	9,889 gals.
Total gallons required to fill Rx. Cavity from el. 703' to 726" 1 1/2".	228,691 gals.
Total gallons required to fill Rx. Cavity (initial fill).	295,362 gals.
Total gallons required to fill Fuel Transfer Canal (initial fill).	102,732 gals.

APPENDIX K
Page 2 of 2

REACTOR CAVITY LEVEL INDICATOR
(Refueling Canal Ladder)



REFUELING CANAL (EQUIPMENT PIT) LADDER
42 RUNGS SPACED AT 1'-0"

13. 035 K5.01 001

Unit One is at 90% power when the S/G #4 safety fails open.

Which one of the following describes how the unit will initially respond?

A. Reactor power will remain the same because the S/G levels are on program.

✓B. The increased steam flow will insert positive reactivity.

C. Reactor power will increase due to a higher Tave.

D. Pressurizer level will increase due to a higher program level.

A. Incorrect, reactor power will change to due a lower Tave, positive reactivity.
B. Correct, increased steam flow will drop Tave which inserts positive reactivity.
C. Incorrect, Tave will not increase; reactor power will not increase since this is a negative reactivity effect.
D. Incorrect, pressure level will drop due to shrinkage when Tave drops and pressurizer level program inputs are auctioneered Tave vs. no-load Tave. Since Tave drops then auctioneered Tave would initially drop.

K/A[CFR]:	035 K5.01 [3.4/3.9] [41.5]
Reference:	AOP-S.05 symptoms.
LP/Objective:	OPL271MS B.9
History:	New question.
Level:	~~Comprehension~~ memory
Comments:	FHW 12/02 035 K5.01

3.1 Symptoms

C. The following automatic actions may occur due to a steam line or feedwater line break or leak:

- Reactor trip on OTΔT.
- Automatic rod withdrawal due to dropping T-avg.
- Automatic rod withdrawal stop from C-3 and C-4.
- Turbine runback from OTΔT or OPΔT.

3.2 Entry Conditions

None

16. 059 A4.01 001

Unit One is operating at 75% power.

Instrument Mechanics are performing a test on the S/G level transmitters. They accidentlly make up a hi-hi level signal on channels II and III for S/G #1.

Which one of the following is correct for this condition?

A. Only 1-FCV-3-33 feedwater isolation valve to S/G #1 will close.

✓B. "Main Feedwater Pump Turbine 1B Abnormal" alarm will annunciate.

C. "Turbine Runback BOP" alarm will annunciate.

D. "PS-2-129 Low NPSH at MFP's" alarm will annunciate.

A. Incorrect, all 4 feedwater isolation valves will close, see 1,2-47W611-3-2.

B. Correct, 2/3 level transmitters made up for 1/4 S/Gs will create a P-14 and trip the MFP turbines (see 47W611-3-2). This annunciator will receive an input from the turbine tripped alarm SER.

C. Incorrect, this alarm will annunicate if MFP trip and load >80%, see 1-AR-M2-A (B-1).

D. Incorrect, this alarm comes in if 100 psid decreasing between MFP intlet and #2 heater shell caused by increasing load or CBP trip. see 1-AR-M3-A (E-1).

K/A[CFR]: 059 A4.01 [3.1/3.1] [41.7]

Reference: 0-GO-5 R27 section 5.1 step [20]
1-AR-M3-B (B-1)
1-AR-M2-A (B-1)
1-AR-M3-A (E-1)
1,2-47W611-2-1
1,2-47W611-3-2

LP/Objective: OPL271COND.FW B.8

History: New question.

Level: Comprehension

Comments: FHW 12/02 059 A4.01

Unit ____________ STARTUP No. _________ Date ___________

5.1 Power Ascension From 30% to 100% (Continued)

[19] **WHEN** approximately 40% reactor power:

[a] **VERIFY** annunciator XA-55-4A, window E-7:

C-20 AMSAC ARMED

is **LIT**. ☐

[b] **CLOSE** the drains on the operating main feedwater pump turbine (**N/A** other pump).

MFPT	DESCRIPTION	HANDSWITCH	POSITION	INITIALS
A	DRAIN VALVES	HS-46-14	CLOSED	________
B	DRAIN VALVES	HS-46-41	CLOSED	________

NOTE With verbal approval from the Operations Superintendent, placing the second main feed pump in service may be deferred until power is approximately 65%. Logic prevents opening the standby MFPT condenser isolation valves if the pump is not reset prior to exceeding 9 million lbs/hr flow on the running pump.

[20] **WHEN** approximately 40 to 45% turbine load, **THEN** **PLACE** the second MFPT in service by performing the following:

[a] **IF** the Operations Superintendent has approved one MFP operation during the power ascension, **THEN**

1. **RECORD** which MFPT is in service.

   ________ MFPT ________

2. **MONITOR** loading of the MFP in service as load is increased. ☐

Unit ____________ STARTUP No. ________ Date __________

5.1 **Power Ascension From 30% to 100%** (Continued)

[b] **WHEN** second MFPT is to be placed in service, **THEN**

PLACE second MFPT in service in accordance with 1,2-SO-2/3-1. ☐

[21] **PERFORM** the following as system parameters permit:

[a] **VERIFY** three (3) Hotwell pumps running. ☐

[b] **VERIFY** two (2) Condensate booster pumps running. ☐

[c] **VERIFY** MFW pump(s) in service (only 1 required if approved by Operations Superintendent). ☐

[d] **VERIFY** two (2) #3 heater drain tank pumps running. ☐

[e] **VERIFY** (1) #7 Heater Drain Tank pump in one service. ☐

[22] **IF** the second #7 heater drain tank pump has not been started, **THEN**

START the second #7 heater drain tank pump in accordance with 1,2-SO-5-3. ________

Source	Setpoint	
SER (Internal) Windows A-2 thru A-7, B-2 thru B-7, C-2 thru C-7, and D-2 thru D-7	N/A	MAIN FEEDWATER PUMP TURBINE 1B ABNORMAL

Probable Causes

1. Thrust bearing wear detector.
2. Low bearing oil pressure (turbine and pump).
3. Main feedwater pump 1B condenser isolation valve not fully open.
4. Low vacuum.
5. Low injection water pressure.

Corrective Actions

[1] REFER to source windows on this panel for corrective actions to be taken.

[2] IF source windows are not lit, THEN
TEST annunciator panel to determine if bulbs good.

References

45N646-1,
45B655-03B-0

TURBINE RUNBACK BOP

Source	Setpoint
SER 83 1. ZS-6-105A or 105B, PS-47-13A, and FS-6-107	1. #3 HDT bypass valve LCV-6-105A or 105B open and turbine load > 85%, and FS-6-107 < 5500 gpm after 10 sec. T.D.
2. Either 1A or 1B MFP tripped and PS-47-13B	2. MFP trip and load > 80%.

Probable Causes

1. High level in #3 heater drain tank with turbine load > 85%.
2. Feedwater pump trip with turbine load > 80%.

NOTE

Runback is accomplished through the EHC System Valve Position Limiter.

Corrective Actions

[1] PERFORM the following for a #3 HDT runback:

[a] IF LCV-6-105A or 105B is open, THEN
VERIFY turbine runback to 80%.

[b] IF any #3 HDT pump(s) trips, THEN
GO TO AOP-S.04, *Condensate or Heater Drains Malfunction* WHILE continuing with this instruction.

NOTE

Due to recent sparger redesigns and/or possible flow obstructions in the #3 HDTP bypass piping certain operator actions must be performed should a secondary side runback occur.

[c] IF #3 HDT flow on [FR-6-107] is greater than 3000 gpm, THEN
MONITOR [FR-6-107] AND GO TO step [e].

NOTE

Position of 6-105A and 105B on XX-6-1, *Heater Drain Bypass to Condenser* located on panel M-2 may be used to determine if #3 HDT level has stablized in the following step. Should 6-105A be FULL open and 6-105B be THROTTLED, then turbine load has been reduced enough to allow stablization. Should 6-105A and 6-105B be FULL open (red light indication only), then #3 HDT level may NOT be stable and may require further turbine load reduction.

[d] IF #3 HDT flow on [FR-6-107] is less than 3000 gpm, THEN
REDUCE turbine load (use valve position limiter) to a value which allows #3 HDT level to stablize (load could be 60% or less)

AND

GO TO AOP-S.04 if a Feedwater Heater String isolates.

CONTINUED

TURBINE
RUNBACK
BOP

CONTINUED

[e] DISPATCH an operator to monitor #3 HDT level.

[f] EVALUATE removal of one #3 HDT pump, and/or one Condensate Booster pump, and/or one or all Condensate Demineralizer Booster pumps to reduce condensate header pressure.

[2] PERFORM the following for a MFP trip:

[a] ENSURE operating MFP loads up and main reg valves respond to control S/G levels.

[b] ENSURE main feedwater controls are returning to a stable position for the present plant conditions.

[c] GO TO AOP-S.01, *Loss of Normal Feedwater.*

[3] REFER TO 0-GO-5, *Normal Power Operations,* to remove turbine control from governor valve limiter.

References 45N647-8, 45B655-02A-0, 45N657-26, 47B601-6-21, 47B601-47-6

Source	Setpoint	
SER 214 PS-2-129A	100 psid decreasing (differential between MFP inlet and #2 heater shell)	**PS-2-129 LOW NPSH AT MFP'S**

Probable Causes

1. Increasing load.
2. Condensate booster pump trip.

Corrective Actions

[1] ENSURE MFP inlet pressure greater than 320 psig by reducing turbine load [M-3, PI-2-129].

[2] ENSURE the following pumps are operating as required by 0-GO-5 for the present unit load.

a. Hotwell pumps
b. Condensate DI booster pumps
c. Condensate booster pumps
d. #7 HDT pumps
e. #3 HDT pump

[3] ADJUST condensate and feedwater pressure as required to clear alarm.

[4] IF MFP trips, THEN GO TO AOP-S.01, *Loss of Normal Feedwater.*

References

47B601-2-21
45B655-03A-2
47W611-2-2

18. 060 AK2.02 001

Waste Gas Decay Tank "A" is in service and has developed a small external leak on the gas analyzing sampling valve. Containment purge & waste gas release are in progress

Which of the following are automatic actions ~~that could~~ would take place as result of this condition?

A. ~~A~~ The waste gas release, ~~if in progress,~~ will terminate.

B. The Containment purge, ~~if in progress,~~ will terminate.

✓C. Auxiliary building ventilation will shut down.

D. Nitrogen to the waste gas system will isolate.

A. Incorrect, RE-118 is a closed pipe detector and not sensing this leak.
B. Incorrect, RM-130/131 has closed pipe sensing and not subject to this leak.
C. Correct, the spent fuel pool monitors or RM-90-101 will initate an ABI that will stop the normal ventilation.
D. Nitrogen has a manual isolation valve and no circuit with rad monitors.

Student must understand the external leak will drift inside the auxiliary building and could make the spent fuel pit ARMs or RM-90-101 reach setpoint and initiate an ABI that will stop the normal auxiliary building ventilation.

K/A[CFR]:	060 AK2.02 [2.7/3.1] [41.7]
Reference:	1,2-47W830-4 1,2-47W611-30-6 0-SO-90-2 0-SO-30-10
LP/Objective:	OPL271RADMON B.9
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 060 AK2.02

3.0 PRECAUTIONS AND LIMITATIONS (Continued)

K. If an A train exhaust fan is going to be used in conjunction with a B train exhaust fan, the non-operating fan to be started needs to be manually set and left in manual for optimum operating performance.

L. Ensure that essential raw cooling water (ERCW) headers are charged up to the safety feature equipment coolers prior to making them operational.

M. The PASF will <u>not</u> be used as a normal sampling facility, but periodic sampling may be performed by the Radiochemical Laboratory. When this is done, radiation levels in the Auxiliary Building should be monitored.

N. After an ABI due to high radiation in the Auxiliary Building, restart of the Auxiliary Building supply and exhaust fans may cause a re-initiation of the ABI. Inform the appropriate Operations personnel prior to restarting the Auxiliary Building Ventilation System.

O. An Auxiliary Building Ventilation Shutdown due to smoke detector cross zone is not an ABI or ESF. It only stops the Supply, Exhaust, and Fuel Handling Fans. Auxiliary Building Ventilation fans will be placed and remain in the **PULL-TO-LOCK** position unit **[<u>1-HS-30-102D</u>]** has been **RESET**.

P. An Auxiliary Building isolation will be initiated from one of the following actuation signals:

1. Phase A containment isolation
2. Radiation detection in the refueling area
3. Auxiliary Building General exhaust vent high radiation
4. Auxiliary Building general supply fan suction high temperature
5. Manual
6. Manual with 1-HS-30-101A or 1-HS-30-101B on 1-M-6.
7. Manual with 2-HS-30-101A or 2-HS-30-101B on 2-M-6.
8. High radiation in the auxiliary building exhaust stack. (0-RM-90-101B, on 0-M-12)
9. Phase A containment isolation, either unit.
10. High radiation in refueling area, 0-RM-90-102 > 10mr/hr. (A Train)
11. High radiation in refueling area, 0-RM-90-103 > 10mr/hr. (B Train)
12. High temperature in auxiliary building general supply fan suction, either unit, > 115°F: 1-TS-30-103 and 103A; 2-TS-30-104 and 104A.

ALIGNMENT OF (0-RE-90-118)

DESCRIPTION	LOCATION	VALVE NO.	REQUIRED POSITION	VERIFIED BY
WGDT Discharge Isolation	Upstream of 0-RE-90-118	0-77-840D	LOCKED OPEN	____ 1st ____ IV
WGDT Discharge Isolation	Downstream of 0-RE-90-118	0-77-840C	LOCKED OPEN	____ 1st ____ IV
Vent Valve	Downstream of 0-77-840D	0-VTIV-77-840G	CLOSED	____ 1st ____ IV
Vent Valve	Upstream of 0-77-840C	0-VTIV-77-840H	CLOSED	____ 1st ____ IV
Spare isolation Valve	Upstream of 0-77-840D/in valve gallery (1-47W600-106)	0-77-269	CLOSED CAPPED	____ 1st ____ IV
Spare isolation Valve	Downstream of 0-77-840C/in valve gallery (1-47W600-106)	0-77-270	CLOSED CAPPED	____ 1st ____ IV

19. 061 AK2.01 001

"0-RA-90-102A Fuel Pool Rad Monitor High Rad" alarm just annunciated.

Which one of the following describes the detector location and the automatic actions associated with this alarm?

A. East of the spent fuel pool on the penetration room wall elevation 749 and results in an ABI (fuel handling area isolation) train "A."

B. The transfer canal elevation 714 and results in ABI (fuel handling area isolation) trains "A" and "B."

C. West of the spent fuel pool on the penetration room wall elevation 749 and results in an ABI (fuel handling area isolation) train "B."

✓D. Near the spent fuel pool water elevation 734 and results in an ABI (fuel handling area isolation) train "A."

A. Incorrect per references.
B. Incorrect per references, this is 20 feet below floor elevation.
C. Incorrect per references.
D. Correct, per 0-AR-M12-B (B-3) and 0-RA-90-102A is associated with "A" train ABI per drawing. Where as 0-RA-90-103A is associated with "B" train.

K/A[CFR]: 061 AK2.01 [2.5/2.6] [41.7]

Reference: 0-AR-M12-B (B-3)
1,2-47W611-30-6
0-SO-30-10 R19

LP/Objective: OPL271C013 B.2

History: New question.

Level: Analysis

Comments: FHW 12/02 061 AK2.01

exceeds K/A to set 3
plausible distractors

Source

SER 760
0-RE-90-102
Retransmitted to U-2
SER 2243

Setpoint

50 mr/hr > 1 second

0-RA-90-102A
FUEL POOL RAD
MONITOR HIGH RAD

Probable Causes

1. High radiation in spent fuel pit area elevation 734.

Corrective Actions

[1] **VERIFY** the following:
   a. Auxiliary Building General Supply and Exhaust and Fuel Handling exhaust isolate (A-Train) (1-M-9).
   b. Auxiliary Building Gas Treatment System starts (1-M-9).

[2] **CHECK** 0-RM-90-102 and 0-RM-90-103 on 0-M-12 to verify alarm.

[3] **IF** high radiation alarm valid, **THEN**
   [a] **ANNOUNCE** "High Radiation at spent Fuel Pool Area" over PA system.
   [b] **NOTIFY** SM.
   [c] **NOTIFY** RADCON.

[4] **IF** B-Train ABI has not actuated from a valid High Radiation condition, **THEN**
   INITIATE manually B-Train Auxiliary Building Ventilation Isolation via **[1-HS-30-101B]** or **[2-HS-30-101B]** (M-6).

[5] **IF** fuel handling in the Spent Fuel Pit is in progress, **THEN**
   REFER TO AOP-M.04, *Refueling Malfunctions.*

[6] **REFER TO** AOP-M.06, *Loss of Spent Fuel Cooling.*

[7] **IF** Auxiliary Building Ventilation Isolation resulted from an invalid ABI signal, **THEN** ,
   REFER to 0-SO-30-10 *Auxiliary Building Ventilation Systems* to recover from ABI.

[8] **EVALUATE** Technical Specifications 3.3.3.1 and 3.9.12.

[9] **INITIATE** Corrective Actions.

[10] **WHEN** conditions return to normal, **THEN**
   RETURN Auxiliary Building Ventilation System to normal in accordance with 0-SO-30-10, *Auxiliary Building Ventilation Systems.*

References

45B655-12B-0,
47W610-90-1

Date__________

8.5 Verification of Auxiliary Building Isolation Actuations (Continued)

N. Cask Loading Area supply isolation damper.

1. [0-FCO-30-129] ☐
2. [0-FCO-30-130] ☐

O. Cask Loading Area exhaust isolation damper.

1. [0-FCO-30-122] ☐
2. [0-FCO-30-123] ☐

NOTE 1 The Auxiliary Building may not be accessible in the post-accident condition.

NOTE 2 In the following step, both dampers will be closed if Train A & B Auxiliary Building Isolation (ABI) signals were received.

[6] **ENSURE** one damper **CLOSED** for each ABI Train actuated, by local observation.

Auxiliary Building Exhaust Vent isolation damper. (Verification by using lights on 1-HS-30-49 and 1-HS-30-55 located on South wall behind "B" Surge Tank, elevation 734)

1. [1-FCO-30-49] ☐
2. [1-FCO-30-55] ☐

[7] **CHECK** Auxiliary Building being maintained at 0.275-inch of water negative pressure: (Panel 1-M-9)

A. [0-PDI-30-148] Aux Bldg Gas Trtmt Sys. ☐

B. [0-PDI-30-149] Aux Bldg Gas Trtmt Sys. ☐

[8] **WHEN** recovery from ABI is to be performed, **THEN**

PERFORM Section 8.4, Recovery From Auxiliary Building Isolation. ☐

End of Section 8.5

20. 061 K2.01 001

Which one of the following is the correct power supply to the Auxiliary Feedwater System ERCW Header isolation motor operated valve 1-FCV-3-136B?

A. 480v Unit Bd 1A.

B. 480v Turbine MOV Bd 1A.

✓C. 480v Rx MOV Bd 1A2-A.

D. 250v Battery Bd #1.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

The student must understand this vital system is fed from a shutdown power source which is 480v Rx MOV Bd 1A2-A.

K/A[CFR]:	061 K2.01 [3.2/3.3] [41.7]
Reference:	1-SO-3-2, Att. 1, date 16 Nov 01, page 6 of 8. 1,2-45N779-10
LP/Objective:	OPL271C035 B.4
History:	New question.
Level:	Memory
Comments:	FHW 12/02 061 K2.01

EQUIPMENT I.D.	PRINT REFERENCE	BREAKER POSITION	FUSES INSTALLED	TRANSFER SWITCH	INITIALS
ERCW Header Isol Valve 1-FCV-3-136B	45N751-3 45N779-10	480V Rx MOV Bd 1A2-A/3B 1-BCTD-3-136B-A CLOSED	Four 1 amp BUSS FRN 1 1-FU4-3-136 B-A	1-XS-3-136B NORMAL	____ 1st ____ IV
ERCW Header Isol Valve 1-FCV-3-179A	45N751-8 45N779-10	480V Rx MOV Bd 1B2-B/11E 1-BCTD-3-179A-B CLOSED	Four 1 amp BUSS FRN1 1-FU4-3-179A-B	1-XS-3-179A NORMAL	____ 1st ____ IV
ERCW Header Isol Valve 1-FCV-3-179B	45N751-8 45N779-10	480V Rx MOV Bd 1B2-B/11B 1-BCTD-3-179B-B CLOSED	Four 1 amp BUSS FRN1 1-FU4-3-179B-B	1-XS-3-179B NORMAL	____ 1st ____ IV
TDAFW Pump Isol Valve 1-FCV-1-15	45N751-4 45N779-28	480V Rx MOV Bd 1A2-A/19A 1-BCTD-1-15-A CLOSED	Four 1 amp BUSS FRN 1 1-FU4-1-15A	1-XS-1-15 NORMAL	____ 1st ____ IV
TDAFW Pump Isol Valve 1-FCV-1-16	45N751-4 45N779-28	480V Rx MOV Bd 1A2-A/17B 1-BCTD-1-16-A CLOSED	Four 1 amp BUSS FRN 1 1-FU4-1-16-A	1-XS-1-16 NORMAL	____ 1st ____ IV
TDAFW Pump dc Manual Transfer Switch, T & T Valve 1-FCV-1-51, and dc Vent Fan Normal Power Supply	45W646-6	125V dc Vital Batt Bd III 1-BKRC-3-KC/321-A CLOSED	Three 6 amp BUSS KWN6 1-FU3-1-52-S (5)	N/A	____ 1st ____ IV

(5) Fuses located in JB 3043, elev 669.

21. 061 K6.01 001

Unit One is performing a surveillance test on the Terry Turbine when the following is experienced.

"Aux FWP Turbine 1A-S Mechanical Overspeed Trip" alarm is lit because 1-FIC-46-57 failed to control pump outlet flow. Controller output is 100% 1-SO-3-2

Which one of the following is the correct operator action(s) to place the pump back in service with the failed controller?

A. Place 1-HC-46-57-S in manual and go to open with 1-HS-1-51A-S.

B. Ensure the mechanical overspeed mechanism is latched and place 1-HC-46-57-S in manual and go to open with 1-HS-1-51A-S.

C. After the overspeed alarm is cleared place 1-HC-46-57-S in manual and go to open with 1-HS-1-51A-S.

✓D. After the overspeed alarm is cleared place 1-HC-46-57-S in manual and set the controller output to 20% and go to open with 1-HS-1-51A-S.

A. Incorrect, per reference.

B. Incorrect, per reference.

C. Incorrect, since the controller failed to control before then adjust the controller output to 20%.

D. Correct, the overspeed condition has been corrected and the controller output has been adjusted to 20% to prevent another overspeed.

K/A[CFR]:	061 K6.01 [2.5/2.8] [41.7]
Reference:	1-AR-M3-C (A-4) 1-SO-3-2 R27 section 8.5 and section 5.3.
LP/Objective:	OPL271C035 B.6
History:	New question.
Level:	Memory
Comments:	FHW 12/02 061 K6.01

Date________

5.3 Startup and Control of TDAFW Pump from the UCR

NOTE 1 Possible cause for TDAFW Pump not manually starting is tripping of thermal overloads for the trip and throttle valve at ITE box next to the local controls in the Terry Turbine Room.

NOTE 2 AFW should be maintained at a constant flow rate to reduce the potential for S/G nozzle cracking. The goal should be to establish the lowest possible constant flow rate.

[1] **ENSURE** AFW system is in Standby per Section 5.1 Startup/Standby Readiness of this Instruction. ________

[2] **NOTIFY** the appropriate operator to locally inspect the TDAFW pump to ensure it is ready for operation. ________

[3] **ENSURE** the following level control valves are **CLOSED**:

VALVE NUMBER	DESCRIPTION	INITIALS
1-LCV-3-172	S/G 3 LCV	______
1-LCV-3-173	S/G 2 LCV	______
1-LCV-3-174	S/G 1 LCV	______
1-LCV-3-175	S/G 4 LCV	______

Date__________

5.3 Startup and Control of TDAFW Pump from the UCR (Continued)

[4] **IF** desiring to control Terry Turbine speed manually, **THEN**

PLACE [1-HC-46-57-S] in **MANUAL** and set controller as desired to control pump speed. ______

[5] **START** the Terry Turbine by PLACING [1-HS-1-51A-S] in the **OPEN** position. ______

[6] **VERIFY** pump comes up to operating speed of ~ 3970 (or desired speed) rpm as seen on [1-SI-46-56A-S]. ______

[7] **CONTROL** S/G level as necessary by PLACING the applicable level control valve handswitch in the **OPEN** position until desired S/G level is achieved, **THEN**

PLACE the applicable level control valve handswitch in the **CLOSED** position. ______

NOTE 1 To increase the pressure of [1-PCV-3-183], turn the valve hand wheel in a clockwise direction. To decrease pressure Turn valve hand wheel in a counter-clockwise direction.

NOTE 2 Due to the varied inlet steam pressures encountered with TDAFW pump operation, [1-PCV-3-183] setpoint must be adjusted to prevent steam seal leakage. A setting of ~ 120 psig corresponds to a 1000 psig inlet steam pressure.

[8] **IF** TDAFWP inlet steam supply pressure is ~ 1000 psig, **THEN**

VERIFY [1-PCV-3-183], Ejector Pressure Control Valve, is controlling pressure at approximately 120 psig as indicated on [1-PI-3-184]. ______

[9] **PERFORM** visual inspection to determine if steam is leaking down the turbine shaft through the gland steam seals. ☐

Date__________

5.3 Startup and Control of TDAFW Pump from the UCR (Continued)

[10] **IF** there is steam leaking through the steam seals, **THEN** adjust **[1-PCV-3-183]**, Ejector Pressure Control Valve to prevent leakage by performing the following:

[a] **REQUEST** MIG to install a temporary pressure gauge (~ 1000 psig range) at PI-3-184 location. ☐

[b] **ADJUST** **[1-PCV-3-183]** to obtain a pressure of ~ 500 psig. (This will clear the PCV of any debris). ☐

[c] **IF** additional pressure control is required, **THEN**

THROTTLE **[1-VLV-3-920]** **WHILE** adjusting **[1-PCV-3-183]**. ☐

[d] **IF** TDAFWP inlet steam supply pressure is ~ 1000 psig, **THEN**

ADJUST **[1-PCV-3-183]** to obtain a pressure of ~ 120 psig. ☐

[e] **IF** TDAFWP inlet steam supply pressure is **NOT** ~ 1000 psig, **THEN**

ADJUST **[1-PCV-3-183]** to prevent leakage of steam seals down the turbine shaft. ☐

[f] **REQUEST** MIG to remove the temporary pressure gauge and to reinstall **[PI-3-184]**. ☐

[g] **ENSURE** **[1-VLV-3-920]** is **OPEN**. ________/ ________
1st IV

END OF TEXT

Date__________

8.5 Operation of TDAFW Pump 1-FCV-1-51 (Trip and Throttle) Valve if it Closes Unintentionally or Will Not Open

[1] **IF** TDAFW pump Trip and Throttle valve will not electrically open or closes unintentionally, **THEN**

ENSURE the following:

[a] **ENSURE** Trip and Throttle Valve is run down to full close position. ______

[b] **ENSURE** mechanical overspeed mechanism is **LATCHED** (refer to local placard). ______

[c] **ENSURE** Terry Turbine overspeed alarm in UCR is clear. ______

[2] **IF** an overspeed trip has occurred due to 1-FIC-46-57 failing to control pump outlet flow automatically, **THEN**

PLACE [1-HC-46-57-S] in **MANUAL** and set controller output to 20 percent as seen on **[1-FI-46-57-S]** on M-3 prior to attempting restart of TDAFW pump. ______

[3] **IF** needing to continue Terry Turbine operation, **THEN**

GO TO applicable startup section. ______

END OF TEXT

Source	Setpoint	
SER 264 SS-46-53	4900 rpm (125%)	**AUX FWP TURBINE 1A-S MECHANICAL OVERSPEED TRIP**

Probable Causes

1. Governor malfunction
   a. Leaky governor or leak near governor.
   b. Governor responds slowly due to worn parts or sticking.
   c. Linkage out of adjustment.
   d. Low governor valve hydraulic fluid.

Corrective Actions

[1] IF auxiliary feedwater pump turbine was in service, THEN
ENSURE feedwater pump turbine tripped.

[2] IF motor driven auxiliary feedwater pumps are not in service and are available, THEN
START additional auxiliary feedwater pumps as needed.

NOTE

To restart the auxiliary feedwater pump turbine from a mechanical overspeed the mechanical trip linkage will have to be reset locally.

[3] REFER TO 1-SO-3-2, Auxiliary Feedwater System, for further guidance on reset and restart.

References

Terry Steam Turbine Instruction Manual VTD DR04 0010
(Cont. # VTM-I075-0250)
45B655-03C-0,
45N646-6,
47W611-3-4

22. 063 A2.01 001

"125V DC Vital Bat Bd 1 Abnormal" alarm is lit.

Which one of the following is the correct cause and action for this condition?

A. An undervoltage condition exists and dispatch operations personnel to adjust the charger output voltage.

A ground exists, B. Clearing the alarm requires adjustment of the red flag (indicator) ground setpoint by dispatching operations personnel to the local board.

C. An overvoltage condition exists and dispatch operations personnel to adjust the charger output voltage.

A ground exists. D. Clearing the alarm will require a local push button reset and additional IM support in the control room at the annunciator panel.

A. Incorrect, an undervoltage does not have input to this alarm. Also the charger output voltage is adjusting by maintenance section.
B. Correct per reference.
C. Incorrect, a overvoltage does not have input to this alarm. Also the charger output voltage is adjusting by maintenance section.
D. Incorrect, the reset is not a push button and IM support is not required.

K/A[CFR]:	063 A2.01 [2.5/3.2] [41.5 43.5]
Reference:	1-AR-M1-C (A-5)
LP/Objective:	OPN218E.007 B.17
History:	New question.
Level:	~~Memory~~ comp
Comments:	FHW 12/02 063 A2.01

125V DC VITAL BAT BD I ABNORMAL

Source

Setpoint

SER 40

1. Fuses and breaker provided with alarm contacts which close when fuse opens or BKR trips. Contacts wired to common column alarm bus for annunciation. — N/A
2. Ground indicator alarm — ±80V

Probable Causes

1. Any battery board breaker tripped.
2. Any fuse blown on fuse column A, B, C or D.
3. Positive or negative ground on DC system above setpoint (± 80V).

Corrective Actions

[1] **DISPATCH** Personnel to 125V DC Vital Battery Bd I **AND CHECK** following :

[a] Ground present greater than setpoint on 125V DC ground detector.

[b] Any breaker(s) **TRIP** on battery board.

[c] Fuse blown on fuse column A, B, C, or D (located in back of panel D).

[2] **IF** ground present on 125V DC Vital Battery Bd, **THEN INITIATE** maintenance.

[3] **ADJUST** red flag setpoint to clear alarm.

NOTE SRO approval must be provided prior to reclosing any tripped breaker.

[4] **IF** any feeder breaker on 125V Dc Vital Battery Bd **TRIP, THEN**

[a] **NOTIFY** Unit SRO of breaker trip.

[b] **EVALUATE** effects of reclosing tripped breaker.

[c] **IF** permission obtained from SRO, **THEN ATTEMPT** to **CLOSE** tripped Breaker.

[5] **IF** breaker will not **CLOSE** or **TRIPS** again **THEN INITIATE** maintenance.

CONTINUED

23. 064 K2.02 001

The Diesel Generator 1 A-A Day Tank Fuel Oil Transfer Pump #1 was running in automatic due to a low level in a 1 A-A Day Tank caused when maintenance pumped out the day tank . The D/G 1 A-A is **NOT** running at this time.

6.9 kv Unit Bd 1B tripped on a ground fault while the D/G 1 A-A Day Tank Fuel Oil Pump #1 was running.

Which one of the following is correct ~~of~~ for this condition?

A. The Day Tank Fuel Oil Pump will continue to run since it is DC powered from the D/G battery.

B. The Day Tank Fuel Oil Pump will stop and after the 1 A-A D/G ties to the 1 A-A 6.9 kv Shutdown Bd~~., a~~ the Day Tank Fuel Oil Pump will start.

C. The Day Tank Fuel Oil Pump will stop and require manual restart after 6.9 kv Shutdown Bd 1 A-A is energized.

D. The Day Tank Fuel Oil Pump will continue to run since it is powered from "B" train shutdown power.

A. Incorrect, the pump's power supply is Diesel Aux Bd 1A1-A, not DC powered.

B. Correct, the pump's power supply is Diesel Aux Bd 1A1-A which will be energized when the D/G 1 A-A ties on to 6.9 kv Shutdown Bd. 1 A-A.

C. Incorrect, the pump was running in auto when the 6.9 unit bd tripped therefore the switch position remains the same and the transfer pump will start again since the day tank level was not increased to normal level.

D. Incorrect, the pump's power supply is Diesel Aux Bd 1A1-A which is fed from the 480v Shutdown Bd 1A1-A.

K/A[CFR]:	064 K2.02 [2.8/3.1] [41.7]
Reference:	45N771-4 15N500
LP/Objective:	OPL271 D/G B.10
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 064 K2.02

25. 076 AK3.05 001

Unit Two is currently at 100% power. Chem Lab reports the RCS activity for Iodine 131 is 480 μci/gm.

Information: 2-RM-90-277 (RCDT).
2-RM-90- 278 (RCDT).
2-RM-90-400A (Shield Bldg Vent).
2-RM-90- 400B (Shield Bldg Vent).

Which one of the following symptoms and procedural actions are correct for Iodine removal?

✓A. Monitor 2-RM-90-277, 2-RM-90- 278, and place mixed beds and cation bed demineralizers in service.

B. Monitor 2-RM-90-400A, 2-RM-90-400B, and divert letdown to the hold up tank.

C. Monitor 2-RM-90-277, 2-RM-90- 278, and isolate normal letdown and place excess letdown in service.

D. Monitor 2-RM-90-400A, 2-RM-90- 400B, and place mixed beds and cation bed demineralizers in service.

A. Correct, RM-90-277/278 monitors for high RCS activity; demin beds reduce RCS activity per references.
B. Incorrect, RM-90-400 monitors only during a gas release; hold up tank does not remove RCS activity.
C. Incorrect, placing excess letdown in service will by pass the demineralizers.
D. Incorrect, RM-90-400 monitors only during a gas release.

K/A[CFR]:	076 AK3.05 [2.9/3.6] [41.5 41.10]
Reference:	1-47W809-1 1-AR-M30-A (A-3) 1-AR-M30-A (B-3) AOP-R.06
LP/Objective:	OPL271C370 B.3
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 076 AK3.05 Written by Scott Poteet.

Source	Setpoint	
SER 1731 1-RE-90-278	316mR/Hr	1-RA-278A RCDT HI RAD

Probable Causes

1. Fuel cladding failure.
2. High activity in reactor coolant.

Corrective Actions

[1] VERIFY [1-FCV-77-10] RCDT pump to TDCT isolation valve closes.

[2] CHECK [1-RM-90-278] for high radiation on panel 1-M-30.

[3] IF need to pump RCDT to FDCT with this high radiation condition present, THEN

REQUEST permission from SRO to place the following hand switches in block:

a. [1HS-77-9B]

b. [1-HS-77-10B]

[4] WHEN ready to pump RCDT, THEN

OPEN [1-FCV-77-10] and [1-FCV-77-9].

[5] REQUEST radiochemical laboratory sample to verify activity.

[6] IF high activity verified, THEN

GO TO AOP-R.06, *High RCS Activity.*

References

47B601-55-75, 47W610-90-4

Source	Setpoint	
SER 1739 1-RE-90-277	316 mR/Hr	1-RA-277A RCDT HI RAD

Probable Causes

1. Fuel cladding failure.
2. High activity in reactor coolant.

Corrective Actions

[1] VERIFY [1-FCV-77-9] RCDT pump to TDCT isolation valve closes.

[2] CHECK [1-RM-90-277] for high radiation on panel 1-M-30.

[3] IF need to pump RCDT to FDCT with this high radiation condition present, THEN
REQUEST permission from SRO to place the following hand switches in block:
  a. [1-HS-77-9B]
  b. [1-HS-77-10B]

[4] WHEN ready to pump RCDT, THEN
OPEN [1-FCV-77-10] and .[1-FCV-77-9].

[5] REQUEST radiochemical laboratory sample to verify activity.

[6] IF high activity verified, THEN
GO TO AOP-R.06, *High RCS Activity.*

References

47B601-55-75, 47W610-90-4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.0 OPERATOR ACTIONS (cont'd)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**MONITOR** Auxiliary Building area radiation monitors NORMAL. • 1 (2)-RR-90-1A and 1B • ICS Area Rad Monitor Displays	**NOTIFY** RADCON of abnormal radiation levels. **USE** PA system to evacuate affected areas.
6.	**PERFORM** the following based on Chem Lab recommendations: a. **PLACE** mixed beds and cation bed demineralizer in service **USING** 1,2–SO-62-9, CVCS Purification System. b. **ADJUST** letdown flow.	
7.	**GO TO** appropriate plant procedure.	

END OF SECTION

29. 103 A3.01 001

Which one of the following is correct concerning an automatic phase B containment isolation?

✓A. Panel 6E will be lit inside & outside and both ~~containment spray pumps will be running.~~

B. Panel 6E will be lit outside the outlined area only ~~and the containment spray pumps will **NOT** be running.~~

C. Panel 6E will be lit inside the outlined area only ~~and the containment spray pumps will **NOT** be running.~~

D. Panel 6E will be dark ~~and both containment spray pumps will be running.~~

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]:	103 A3.01 [3.9/4.2] [41.7]
Reference:	E-0 R23 step 9
LP/Objective:	OPL271C379 B.3
History:	New question.
Level:	Memory
Comments:	FHW 12/02 103 A3.01

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**MONITOR** containment spray NOT required: • Phase B NOT ACTUATED **AND** • Containment pressure less than 2.81 psid.	**PERFORM** the following: a. **ENSURE** containment spray INITIATED: 1) Containment spray pumps RUNNING. 2) Containment spray header isolation valves FCV-72-39 and FCV-72-2 OPEN. 3) Containment spray recirculation valves to RWST FCV-72-34 and FCV-72-13 CLOSED. 4) Containment spray header flow greater than 4750 gpm per train. 5) Panel 6E LIT. b. **ENSURE** Phase B valves CLOSED: • Panel 6K PHASE B GREEN. • Panel 6L PHASE B GREEN. c. **STOP** RCPs. d. **MONITOR** containment air return fans: 1) **RECORD** present time. __________ 2) **WHEN** 10 minutes have elapsed, **THEN** **ENSURE** containment air return fans are running.

31. 2.1.24 001

This question has reference material attached.

Using ~~drawing~~ the attached drawing which one of the following will cause an idle Hotwell Pump Motor to start?

A. HS-2-33A placed in start, HS-2-33B in RESET and ~~both~~ Either MFP "A" ~~and~~ OR "B" condenser isolation valves closed.

B. HS-2-33A in P-Auto and both MFP "A" and "B" condenser isolation valves in the full open position.

✓C. ~~Both~~ Either MFP "A" ~~and~~ OR "B" condenser isolation valves in the full open position, all electrical faults reset, and HS-2-33B placed in the TEST position.

D. Reset of electrical fault relays while both MFP "A" and "B" condenser isolation valves are open, HS-2-33B in reset and HS-2-33A in spring return to mid position.

A. Incorrect per reference. No flow path if the valves are closed, interlock.

B. Incorrect per reference. No P-Auto position.

C. Correct per reference.

D. Incorrect per reference switch was never in start postion.

K/A[CFR]:	2.1.24 [2.8/3.1] [41.10 43.5]
References:	1,2-47W611-2-1
LP/Objective:	OPN218SS.002 B.6
History:	New Question
Level:	Analysis
Comments:	FHW 12/02 2.1.24 Written by Jim Kearney (10/3/02). **Must provide a copy of drawing 1,2-47W611-2-1**

32. 2.2.4 001 ✓

This question has reference material attached.

Given the following plant conditions:
- Unit One and Unit Two are both in Mode 5.
- Unit One RCS level is being maintained at elevation 708 ft.
- Unit Two RCS level is being maintained at elevation 706 ft.

Which one (1) of the following describes the corresponding RVLIS upper range indications for both units?

	Unit One	Unit Two
✓A.	99%	94.5%
B.	94.5%	94.5%
C.	94.5%	99%
D.	99%	99%

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]: 2.2.4 [2.8/3.0] [45.1/13]

Reference: 0-GO-13 R23 Appendix J.

LP/Objective: OPL271RVLIS B.6

History: New question.

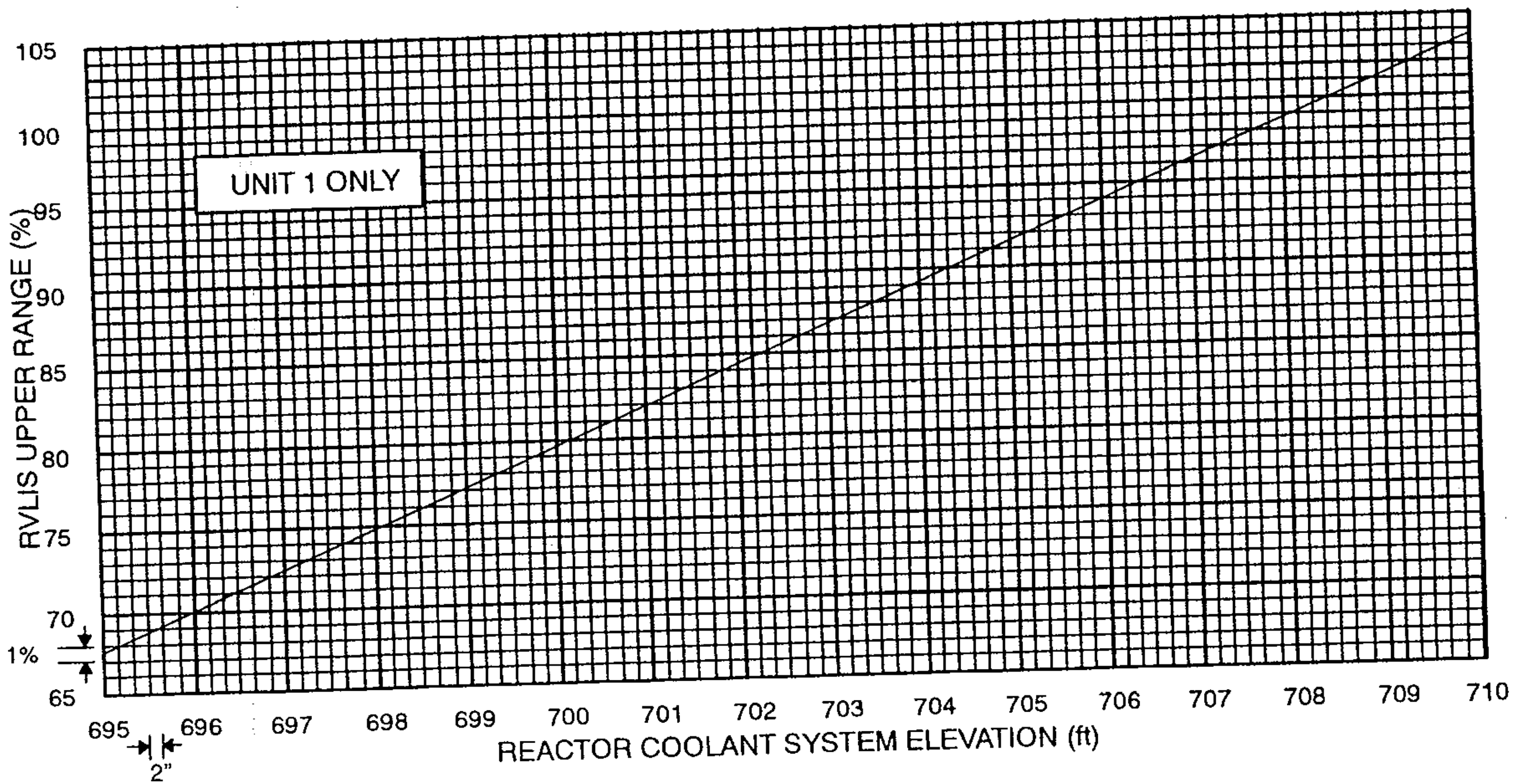
Level: Analysis

Comments: FHW 12/02 2.2.4 Written by Scott Poteet (9/11/02)
Provide a copy of 0-GO-13 R23 Appendix J

APPENDIX J
Page 1 of 2

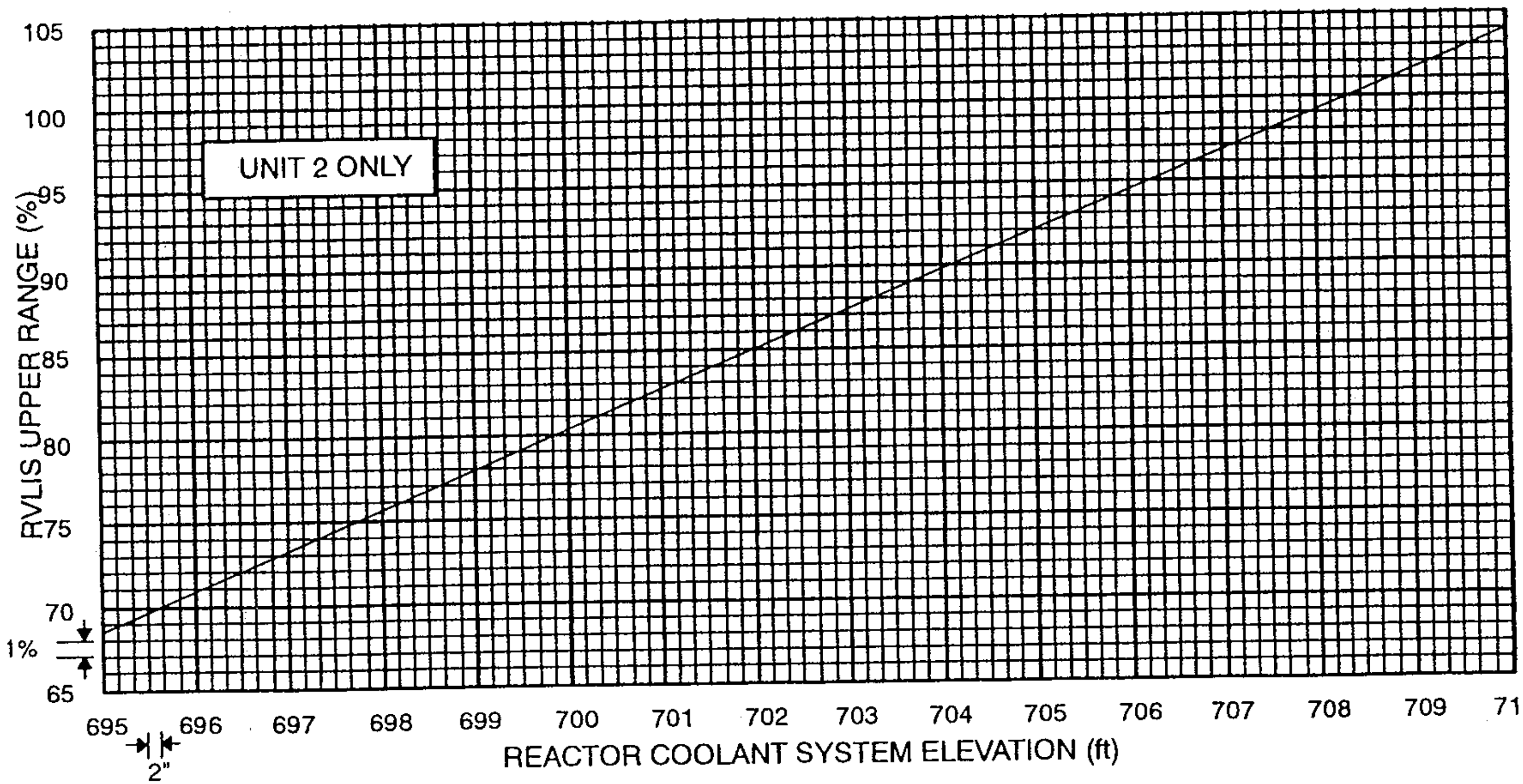
RVLIS UPPER RANGE VS RCS ELEVATION

1. RVLIS head sensor bellows drained/vented and reactor head vented.
2. RVLIS head sensor bellows filled and valved to RX head - add 5 inches.



RVLIS UPPER RANGE VS RCS ELEVATION

1. RVLIS head sensor bellows drained/vented and reactor head vented.
2. RVLIS head sensor bellows filled and valved to RX head - add 5 inches.



34. AIR-B.12 002

Given the following:

- Unit 1 and 2 are at 100% power
- A leak develops on the Control Air System
- Air pressure is at 73 psig

Which ONE ~~(1)~~ of the following describes the system response?

✓A. A and B Auxiliary Air Compressors automatically started and loaded.

B. A and B Train Auxiliary Air automatically isolated from Control Air.

C. A and B Train Auxiliary Air automatically isolated to Containment.

D. A and B Control and Service Air Compressors started and are operating half loaded.

A. Correct, Aux Air Cmpr will start <77 psig and load <83.5 psig.
B. Incorrect this setpoint is <69 psig.
C. Incorrect this setpoint is <50 psig.
D. Incorrect C&S air cmprs will operate at full load at <94 psig.

K/A:	078 K4.01	[2.7/2.9]	
	078 K1.01	[2.8/2.7]	[41.2-9]

Reference: AOP-M.02, Control Air system description.

Objective: OPL271CSA, B.12

History: System bank

Level: Comprehension

Note: FHW 12/02 078 K1.01

D 078 A 3.01 Better match
DO NOT HAVE SENSOR AIR

3.1 Symptoms (cont'd)

A. Deviations or unexpected indications on any of the following may indicate a loss of Control Air System pressure:

- Aux Control Air Header pressure dropping:
  - 0-PI-32-104A, Aux Control Air Hdr A Press [M-15].
  - 0-PI-32-105A, Aux Control Air Hdr B Press [M-15].
- Control and Service (C&S) air compressors tripped.
- Valves or dampers moving to failed position(s).
- S/G anomalies due to MFW Reg Valves failing closed.
- MSIVs fail closed.

B. Any of the following automatic actions may indicate a loss of Control Air System pressure:

SETPOINT (psig)	RANGE (psig) min	RANGE (psig) max	INITIATING EVENT	EVENT
N/A	--	--	Normal control circuit failure	C&S air compressors auto start with backup control circuit
88	86	90	Control Air Receiver dropping	Service Air isolates from Control Air (0-PCV-33-4) C&S air compressors load to 50%
86	84	88	Control Air Receiver pressure dropping	C&S air compressors load to 100%
77	74.5	79.5	Aux Air Receiver pressure dropping	Aux Air Compressors start
69	66.5	71.5	Control Air header pressure dropping	Aux Air isolates from Control Air (0-FCV-32-82 and 85)
50	--	--	Aux Air Receiver pressure dropping	Aux Air to containment valves fail closed. (1-FCV-32-80, -102, -110, 2-FCV-32-81, -103, -111)

3.2 Entry Conditions

None

35. AIR-B.5 014

Given the following plant conditions:

- Control Air receiver #1 pressure is 96 psi and steady
- Control Air receiver #2 pressure is 99 psi and steady
- Service air receiver pressure is 78 psi and decreasing

Which ONE ~~(1)~~ of the following ~~describes the~~ could cause the ~~of~~ decreasing service air receiver pressure?

A. Loss of compressors A & B sequencer power.

B. Stuck open blowdown valve on air dryer A tower # 1.

C. Pressure control valve 0-PCV-33-4 failed open.

✓D. Loss of power to pressure control valve 0-PCV-33-4.

A. Control air receivers 1 and 2 pressure is normal thus ruling out an air compressor problem. the difference between control air receiver pressure is calibration tolerances.

B. The air dyers are on the control air header and the capacity of the blowdown valve is within the capacity of the compressors. A stuck open blowdown valve would have to decrease control air receiver pressures down to < 78 psig for this to affect the service air system via PCV-33-4.

C. Pressure control valve 0-PCV-33-4 fails closed on loss of air or loss of power. If the valve could fail open it will not cause the service air reciever to depressurize.

D. O-PCV-33-4 is the crosstie valve that supplies service air from the control air receivers 1 and 2. This valve fails closed on loss of power and closure of this valve will cause service air receiver pressure to decrease while control air pressure remains normal.

35. AIR-B.5 014

K/A{CFR}: 079 K1.01 [3.0/3.1] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}

079 K2.01 [2.3/2.3] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}

079 K4.01 [2.9/3.2] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}

079 A2.01 [2.9/3.2] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}

078 K1.02 [2.7/2.8] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}

References: AOP-M.02 1,2-47W611-32-1
45N632
45N779-6 45N779-18

LP/Objectives: OPL271C038 Obj. B.2, B.5

History: System bank

Level: Comprehension

Comments: LP-5/2000. C&S Air 002.; FHW 12/02 079 K4.01

3.1 Symptoms (cont'd)

A. Deviations or unexpected indications on any of the following may indicate a loss of Control Air System pressure:

- Aux Control Air Header pressure dropping:
  - 0-PI-32-104A, Aux Control Air Hdr A Press [M-15].
  - 0-PI-32-105A, Aux Control Air Hdr B Press [M-15].
- Control and Service (C&S) air compressors tripped.
- Valves or dampers moving to failed position(s).
- S/G anomalies due to MFW Reg Valves failing closed.
- MSIVs fail closed.

B. Any of the following automatic actions may indicate a loss of Control Air System pressure:

SETPOINT (psig)	RANGE (psig)		INITIATING EVENT	EVENT
	min	max		
N/A	--	--	Normal control circuit failure	C&S air compressors auto start with backup control circuit
88	86	90	Control Air Receiver dropping	Service Air isolates from Control Air (0-PCV-33-4)
				C&S air compressors load to 50%
86	84	88	Control Air Receiver pressure dropping	C&S air compressors load to 100%
77	74.5	79.5	Aux Air Receiver pressure dropping	Aux Air Compressors start
69	66.5	71.5	Control Air header pressure dropping	Aux Air isolates from Control Air (0-FCV-32-82 and 85)
50	--	--	Aux Air Receiver pressure dropping	Aux Air to containment valves fail closed. (1-FCV-32-80, -102, -110, 2-FCV-32-81, -103, -111)

3.2 Entry Conditions

None

36. AOP-C.01-B.1 002
Given the following plant conditions:

- The unit is initially operating at 75% power.
- The following symptoms occur:
  - T_{avg} greater than T_{ref}
  - Pressurizer spray valves partially OPEN
  - Pressurizer level INCREASING

Which ONE ~~(1)~~ of the following would cause the above symptoms to occur?

A. Turbine Runback

✓B. Uncontrolled rod withdrawal

C. S/G Safety Valve failed OPEN

D. Power Range Channel N-43 failed high

A. Incorrect, pressurizer level would be above program but decreasing if a runback occured.
B. Correct, pressure increase and water expansion due to increasing RCS temperature from 75% power.
C. Incorrect, S/G safety failing open would decrease Tave.
D. Incorrect, no inputs to pressure control.

K/A: 001 AK1.03 [3.9/4.0] {41.8 41.10}

Reference: AOP-C.01, Symptoms and Entry Conditions

LP/Objective: OPL271AOPC01, B.1

Level: Comprehension

History: Procedure Bank

Comments: FHW 12/02 001 AK1.03

3.0 SYMPTOMS AND ENTRY CONDITIONS

3.1 Symptoms

A. Any of the following annunciators may indicate a rod control system malfunction:

PANEL XA-55-4B, NIS/ROD CONTROL	
A-7	ZB-412A ROD CONTROL BANKS LIMIT LOW
A-6	ROD CONTROL SYSTEM URGENT FAILURE
B-3	NIS POWER RANGE UPPER DETECTOR HI FLUX DEVN OR AUTO DEFEAT
B-7	ZB-412B ROD CONTROL BANKS LIMIT LOW-LOW
C-3	NIS POWER RANGE LOWER DETECTOR HI FLUX DEVN OR AUTO DEFEAT
C-7	ZB-412C BANK D ROD WITHDRAWAL LIMIT HIGH
D-4	COMPUTER ALARM ROD DEV AND SEQ NIS PWR RANGE TILTS
D-7	FULL LENGTH RODS AT BOTTOM
E-3	NC-46B NIS POWER RANGE CHANNEL DEVIATION

PANEL XA-55-5A, REACTOR COOLANT - STM - FW	
C-6	TS-68-2P/Q REAC COOL LOOPS T REF T AUCT HIGH-LOW

PANEL XA-55-6A, REACTOR PROTECTION AND SAFEGUARDS	
B-1	NC-41U/NC-41K NIS POWER RANGE HIGH NEUTRON FLUX RATE

B. Any of the following parameters may indicate rod control system failure:

1. Unwarranted rod motion as indicated by RPI or step counters.
2. Steam header pressure deviations without corresponding turbine load change.
3. T-avg. - T-Ref deviations.

3.2 Entry Conditions

None

44. AOP-R.01-B.2 003

Given the following plant conditions:

- The operating crew has identified a S/G tube leak.
- AOP-R.01 has been implemented.
- Letdown flow = 75 gpm.
- 1 CCP is in service with charging valves FCV-62-93 and 89 full open.
- Charging flow = 160 gpm.
- Pressurizer level is stable at 58%.
- All other parameters are normal

Which ONE (1) of the following best estimates the total primary to secondary leak rate?

A. 55 gpm.

✓B. 75 gpm.

C. 85 gpm.

D. 150 gpm.

a. Incorrect - Nearest if Total letdown plus seal return are used. (160 - 87 -20 =53)
b. Correct - Nearest if correct numbers are used (160 - 87 = 73.)
c. Incorrect - Nearest if only letdown number is included. (160 - 75 = 85)
d. Incorrect - Nearest if letdown flow is neglected and seal return flow is used. (160 - 12 = 148)

K/A {CFR}: APE 037 2.1.7 [3.7/4.4] [43.5]

References: OPL271CVCS

LP/Objectives: OPL271C366 B.2
OPL271C022 B.3

History: Procedure Bank

Level: Analysis

Comments: FHW 12/02 037 2.1.7 **Provide a calculator**

Reactor coolant purification is used to remove the following from the RCS:

1. fission and activation products (in ionic or particulate form);
2. corrosion agents (O_2, chlorides, and fluorides);
3. excess lithium for pH control.

The bases for the purification portion of CVCS design are:

1. protection of RCS integrity and components from chemical attack;
2. permits access to lines carrying reactor coolant (for chemical monitoring);
3. activity release reduction due to leaks.

Various chemicals are added for corrosion control. Chemical addition purposes are:

1. Control pH during startup (Lithium);
2. Scavenge oxygen during startup (Hydrazine);
3. Control oxygen generated by radiolysis (H2 in VCT);

The system is capable of maintaining all reactor coolant chemistry limits.

Seal water injection supplies filtered seal water to reactor coolant pumps as specified by their design. (See the Rector Coolant Pump lesson material for specific operational information.)

Hydrostatic testing - capable of supplying water at maximum test pressure to verify RCS integrity.

Emergency Core Cooling System interrelation
The centrifugal charging pumps and associated valves serve as part of emergency core cooling. The remainder of system is isolated during accident. Depending on the function being lined up for operation or the function being isolated, the SI signal or the θA isolation signal may align the CVCS system components.

System and Component description

Letdown Charging Seal Water, Chemical Control Purification and Make-up

General Description

This portion of the CVCS maintains programmed pressurizer level by balancing letdown flow with charging and seal water flow, provides filtered seal water to reactor coolant pump seals and controls reactor coolant system pressure during solid plant pressure control operations. The capability of maintaining system chemistry limits and make-up to the RCS is also provided by the following components and flowpaths.

Flow balance (normal operation):

1. Letdown flow = 75 gpm
2. Seal return flow = 12 gpm
3. Charging flow = 55 gpm
4. Seal injection flow = 32 gpm

Therefore, 75 gpm letdown flow + 12 gpm seal return flow = 87 gpm, 55 gpm charging flow + 32 gpm seal injection flow = 87 gpm.

45. AOP-R.01-B.2 004

Given the following plant conditions:

- The plant is operating at 90% power.
- A tube leak has developed on SG #1.
- PRESSURIZER LEVEL HIGH-LOW annunciator has just alarmed.
- Pressurizer level is below program, decreasing slowly.

Which ONE ~~(1)~~ of the following actions is required for these conditions in accordance with AOP-R.01, "Steam Generator Tube Leak"?

A. Manually trip the reactor and actuate SI.

B. Isolate letdown and evaluate pressurizer level.

✓C. Maximize charging for pressurizer level evaluation.

D. Maximize VCT blended makeup to prevent approach to RWST swapover level.

A. Incorrect, not procedurally required with initial conditions.
B. Incorrect, not procedurally required with initial conditions.
C. Correct per procedure.
D. Incorrect, not procedurally required with initial conditions.

K/A {CFR}:	037 2.4.49	[4.0/4.0]	
	037 AA1.11	[3.4/3.3]	{41.7}
	2.4.4	[4.0/4.3]	41.10, 43.2
	2.4.11	[3.4/3.6]	41.10, 43.5

References: AOP-R.01

LP/Objectives: OPL271C366 B.1

History: Procedure Bank

Level: Memory

Comments: Reviewed by J. Epperson; FHW 12/02 037 AA1.11

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Rising Secondary Radiation and Low Pressurizer Level

NOTE: This section should be used if tube leakage has an observable effect upon charging flow and/or Pressurizer level OR if directed by Section 2.2 due to leak rate exceeding 75 gallons per day with an increasing rate of leakage greater than 15 gpd/30 minutes.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CONTROL** charging flow as necessary to maintain Pressurizer level on program.	
2.	**MONITOR** Pressurizer level STABLE or RISING.	**IF** loss of Pressurizer level is imminent, **THEN** **PERFORM** the following: a. **TRIP** the reactor. b. **INITIATE** Safety Injection. c. **GO TO** E-0, Reactor Trip or Safety Injection.
3.	**MAINTAIN** VCT level greater than 13% using automatic or manual makeup.	**IF** VCT level CANNOT be maintained, **THEN** **ENSURE** CCP suction aligned to RWST.

46. AOP-R.02-B.2 001

Unit 1 is at 100% RTP when a small reactor coolant system leak develops. The operators are responding to the event attempting to locate the leak. The following parameters are observed:

- RCS pressure at approximately 2205 psig and decreasing
- All ice condenser doors open
- Containment pressure at 1.5 psid and increasing
- Pressurizer level at approximately 55% and decreasing (with Maximum charging)

Which ONE (1) of the following actions should the operating crew perform?

A. Initiate Phase A Containment Isolation.

B. Initiate Phase B Containment Isolation.

C. Initiate rapid load decrease per AOP-C.03.

✓D. Trip the reactor and initiate Safety Injection.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference, this is a management expectation. Ice doors are open and containment pressure is approaching SI setpoint of 1.54 psid.

K/A:[CFR]: 2.4.4 [4.0/4.3] [41.10 43.2]

Reference: EPM-4 R12 3.4 B. 2 (Management expectation)

LP/Objective: OPL271C266 B.14

History: Procedure bank, old Bank Number B-0383A

Level: Comprehension

Comments: FHW 12/02 2.4.4

3.4 Management Expectations

A. When taking prudent operator actions, the operator is expected to:

1. Stabilize the plant.
2. Utilize supporting procedures as appropriate.

B. When a parameter is approaching a protective action setpoint in an uncontrolled manner, the operator is expected to:

1. Evaluate the parameter magnitude and trend.
2. Take actions as necessary to place the plant in a safe condition without relying solely on automatic actions.

C. Crew briefs are expected to meet the following guidelines:

1. Crew briefs should occur at each major procedure transition, except when implementing RED or ORANGE path FRPs. If implementing a RED or ORANGE path FRP, no brief should be conducted; the SM or procedure reader should merely state that a RED or ORANGE path condition exists and identify the procedure being implemented. If a RED or ORANGE path condition is identified during a brief, the brief should immediately be terminated and the appropriate FRP implemented
2. Provided that no RED or ORANGE path condition exists, briefs may also be conducted at other times (determined by the SM) if necessary to ensure all crew members are cognizant of current plant status or mitigative actions.
3. The brief should consist of all mitigating crew(1) members, except for the brief held at the first transition from E-0. When the step is reached to transition from E-0, the person selected to monitor status trees will begin monitoring status trees and will NOT attend in the brief.
4. Briefs may be deferred or waived by the SM depending on plant conditions and whether the operating crew is aware of the current situation.

D. When two or more procedures must be performed concurrently, the SM is expected to determine the highest priority procedure and ensure the mitigating crew is focused on that procedure.

(1) Defined in Section 5.0, Definitions.

49. AOP-R.05-B.5 001

Given the following plant conditions:

- Unit 1 is operating at 100% power.
- CCS surge tank level was increasing but is now stable.
- "1-RA-90-123A CCS LIQ EFF MON HIGH RAD" Alarm is LIT.
- Surge tank vent valve is closed.

automatic actions that shall take place

Which ONE ~~(1)~~ of the following describes the ~~control board indications that are~~ consistent with the above conditions?

The thermal barrier containment isolation inlet and outlet valves:

A. to the affected RCP close (the non-affected RCPs isolation valves remain open), and the thermal barrier booster pumps trip.

B. to the affected RCP close (the non-affected RCPs isolation valves remain open), the inlet and outlet CCS valves to the RCP oil coolers close, and the thermal barrier booster pumps continue to run with miniflow valves open.

✓C. to all 4 RCPs close, the inlet and outlet CCS valves to the RCP oil coolers remain open, and the thermal barrier booster pumps trip.

D. to all 4 RCPs close, the inlet and outlet CCS valves to the RCP oil coolers are **NOT** affected, and the thermal barrier booster pumps continue to run with miniflow valves open.

A. RCP thermal barriers do not have pump specific containment isolation valves. CIVs to all RCPs will close and thermal barrier pumps will trip.
B. CIVs for RCP thermal barriers will close, but CIVs for RCP oil coolers will not close. The thermal barrier pumps will trip--they do not have miniflow valves.
C. Correct:
D. The first two parts are correct, however thermal barrier pumps will trip--they do not have miniflow valves.

these are not control board indications but listing of automatic actions which may not have indications,

49. AOP-R.05-B.5 001

K/A{CFR}:	008 K1.04	[30./3.3]	{41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}
	008 K6.04	[2.1/2.3]	{41.7}
	008 A3.02	[3.2/3.2]	{41.7}
	009 EK3.01	[3.1/3.6]	{41.5, 41.10}
	009 EK3.05	[3.4/3.8]	{41.5, 41.10}
	009 EK3.09	[3.1/3.4]	{41.5, 41.10}
	015 AK2.08	[2.6/2.6]	{41.7}

References: 0-AR-M12-A (B-1)
47W611-70-3

LP/Objectives: OPL271C026 Obj. 5, OPL271C026 Obj. 12, OPL271C026 Obj. 9, OPL271C367 Obj. 5

History: Modified SYS.bnk Q# CCS 007 stem and 2 distractors
Not modified for 12/02

Level: Memory

Comments: LP-5/2000. CCS 001
FHW 12/02 015AK2.08

No loss of RC Flow

what are auto actions for Hi Temp monitor

This is more likely CCWS K/A

Source	Setpoint	
SER 718 1-RM-90-123A CCS Hx outlet activity monitor 1A1 or 1A2	Refer to TI-18 for setpoint calculation	**1-RA-90-123A CCS LIQ EFF MON HIGH RAD**

Probable Causes

1. RCS, CVCS, RHR or Primary Sampling system leakage to component cooling water system. **[C.3]**
2. Performing test on monitor.

Corrective Actions

[1] **CHECK** instruments on 0-M-12 (1-RM-90-123A and 1-RR-90-123) to determine if alarm is due to high radiation. **[C.3]**

[2] **IF** alarm is due to high radiation, **THEN**

[a] **ENSURE** 1-FCV-70-66 and 2-FCV-70-66 Component Cooling Water Surge Tank Vents are **CLOSED**. **[C.3]**

[b] **IF** CCS to RCP thermal barrier cooling coils has isolated, **THEN**

GO TO AOP-M.03, *Loss of Component Cooling Water,* for loss of Component Cooling Thermal Barrier Booster Pump(s). **[C.3]**

[c] **IF** source of in-leakage is unknown, **THEN**

REFER TO AOP-R.05, RCS Leak and Leak Source Identification to locate, **AND**

ISOLATE source of leak. **[C.3]**

[d] **NOTIFY** Radiochemical Laboratory to sample component cooling water for radioactivity. **[C.3]**

[e] **NOTIFY** RADCON to survey for radiological hazards. **[C.3]**

[f] **CONSIDER** system "feed and bleed" in accordance with 1-SO-70-1, *Component Cooling Water System Train A,* to reduce radiation levels. **[C.3]**

References

45B655-12A-0
47B601-90-38
47W610-90-2

50. CCS-B.9.A 001

Given the following plant conditions:

- Unit #1 is operating at 100%.
- All systems aligned normal.
- Loss of ERCW Supply header 2A occurs due to a rupture in the yard.

Which ONE ~~(1)~~ of the following describes indications the Unit 1 operator would see in the main control room in this event? (Assume no operator actions).

A. Ice condenser chillers trip.

B. "Diesel Gen 1 A-A Jacket Water Temp High-Low Engine 1 or 2" alarm lit on the non running 1 A-A D/G.

C. General ventilation chillers trip.

✓D. CCS surge tank level increasing with auto makeup valve closed.

A. Incorrect - these chillers use RCS even though located in the Aux. Building.
B. Incorrect - ERCW is isolated on a non running D/G.
C. Incorrect - Aux Building chillers are also RCW.
D. Correct - loss of cooling to the A CCS heat exchanger would cause a heatup of A CCS train and expansion into the surge tank.

K/A {CFR}:	APE 062 AA1.05[3.1/3.1] 008 K1.01 3.1/3.1 008 K4.02 [2.9/2.7] [41.7]	{41.7, 45.5, 45.6}
References:	0-47W859-1-4 CCS Flow 0-47W845-2 ERCW Flow 0-AR-M26-A (C-4) AOP-M.01 App. C	
LP/Objectives:	OPL271CCS, B.9.a, B.12	
History:	System bank	
Level:	Analysis	
Comments:	FHW 12/02 008 K4.02	Not a design feature or interlock

APPENDIX C

AFFECTED EQUIPMENT LIST (HEADER 2A)

1. Centrifugal Charging Pump 2A Oil Cooler *(Note 1)*
2. Safety Injection Pump 2A Oil Cooler *(Note 1)*
3. CCS Heat Exchangers 1A1/1A2 and 2A1/2A2 *(Note 2)*
4. Lower Containment Coolers 2A and 2C *(Note 3 & 4)*
5. Upper Containment Coolers 2A and 2C *(Note 3 & 4)*
6. Control Rod Drive Vent Coolers 2A and 2C *(Note 3 & 4)*
7. Instrument Room Cooler 2A *(Note 3 & 4)*
8. Space and room cooling equipment *(Note 4)*
9. Containment Spray Heat Exchanger 2A *(Note 5)*
10. Turbine AFW Pump 2A-S Emergency Suction *(Note 5)*
11. Motor Driven AFW Pump 2A Emergency Suction (if return header A is out-of-service) *(Note 5)*
12. Unit 2 RCP Motor Coolers 1 and 3 *(Note 6)*

Note 1: Damage is imminent unless equipment is stopped.
Note 2: Monitor RCPs for TRIP criteria. Multiple systems affected.
REFER TO AOP-M.03, Loss of Component Cooling Water.
Note 3: Containment parameters may increase due to inadequate cooling.
Note 4: Start redundant equipment and secure affected components.
Note 5: ERCW not normally required.
Note 6: **REFER TO** AOP-R.04, Reactor Coolant Pump Malfunctions, if alarming.

DIESEL GEN 1A-A JACKET WATER TEMP HIGH-LOW ENGINE 1 OR 2

Source
SER 918

	Setpoint
1. Engine 1A1	
0-TS-82-5006/1 Low water jacket temp switch	100°F (±3°F)
0-TS-82-5005/1 High water jacket temp switch	186°F (±4°F)
2. Engine 1A2	
0-TS-82-5003/1 Low water jacket temp switch	100°F (±3°F)
0-TS-82-5002/1 High water jacket temp switch	186°F (±4°F)

Probable Causes

1. Malfunction of the thermostatic bypass valve on the cooling system.
2. Malfunction of immersion heaters (shutdown situation).
3. Insufficient heat removal due to dirty cooler, loss of ERCW supply.

Corrective Actions

[1] DISPATCH personnel to D/G Bldg to verify alarm and to monitor D/G water jacket (TI-82-5006/1 and 5003/1) and lube oil temperatures (TI-82-5010/1 and 5008/1).

[2] IF D/G running, THEN

[a] ENSURE [1-FCV-67-68] or [1-FCV-67-66] OPEN, and supplying approximately 650 gpm (1-FI-67-69 or 1-FI-67-277).

[b] REDUCE the D/G 4.4MW rating by .044MW for every 3.5°F the outside ambient air temperature is >90°F.

[c] IF jacket water temperature exceeds 195°F, THEN REDUCE D/G load (MW) until jacket water temperature stabilizes below 195°F.

NOTE

Diesel Generator jacket water temperature from engine shutdown limit is >205°F

[d] EVALUATE need to emergency stop the D/G AND impact on Technical Specification 3.8.1.1 or 3.8.1.2.

[3] IF D/G aligned for standby operation, THEN CHECK water jacket temperatures 115-125°F and lube oil temperatures 90°-120°F.

[4] IF D/G in standby and water jacket temperature less than 100°F or lube oil temperature less than 90°F, THEN

[a] ENSURE [1-FCV-67-66] and [1-FCV-67-68] CLOSED, (ERCW to heat exchangers).

[b] CHECK engine water immersion heaters operable by ensuring immersion heaters aligned in accordance with 0-SO-82-1, *Diesel Generator 1A-A,* Power Availability Checklist.

[5] IF lube oil temperature decreases to 85°F, THEN PERFORM local D/G run in accordance with 0-SO-82-1 to increase oil temperature.

[6] IF lube oil temperature less than 85°F, THEN DECLARE D/G inoperable.

[7] EVALUATE impact on Technical Specifications (3.8.1.1 or 3.8.1.2).

References

45N767-2, 45N767-6, 45N767-9, 45N767-10, 45B655-26A, A950F02501

KA mem

51. CONTROL*RODS 020

At beginning of Core life, the differential control rod worth is larger near the center of the core compared to the top and bottom of the core.

Which ~~one~~ AM of the following explains this condition?

A. boron concentration.

B. xenon concentration.

✓C. neutron flux distribution.

D. reactor coolant temperature.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference, neutron flux density is greater at the core center.
D. Incorrect per reference.
K/A[CFR]: 001 K5.30 [2.9/3.1] [41.5]

Reference: Rx Theory- Control Rods - LP1- PR05lr2

LP/Objective: Rx Theory- Control Rods - LP1- PR05lr2 Obj. 5

History: Genfunme bank

Level: Memory

Comments: FHW 12/02 001 K5.30 —— Question has nothing to do with effects of fuel burnout on reactivity

GFES Question

INSTRUCTOR GUIDE

KEY POINTS, AIDS, QUESTIONS/ANSWERS

Example 5-2 / TP 5-17 through TP 5-18

A control rod is positioned in reactor with following neutron flux parameters:

Core average thermal neutron flux = 1×10^{12} neutrons/cm^2 sec.

Control rod tip neutron flux = 5×10^{12} neutrons/cm^2 sec.

If control rod is slightly withdrawn such that tip of control rod is located in a neutron flux of 10^{13} neutrons/cm^2 sec, what effect will this have on differential control rod worth? (Assume average flux is constant.)

Solution:

$$DRW \approx \left(\frac{\phi_{tip}}{\phi_{avg}}\right)^2$$

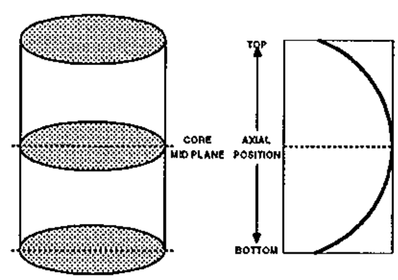
The reactivity worth of tip of control rod can be assumed to be proportional to square of neutron flux that it is in

The increase in neutron flux at tip from 5×10^{12} to 1×10^{13}, which is an increase by a factor of two, produces a DRW increase by a factor of four.

8. As control rod moves in reactor core, differential worth of rod changes
   a. The neutron flux in bare homogeneous core is greatest near midplane of core. Figure 5-5 shows this axial flux variation
   b. Differential rod worth will be largest near core midplane, and lowest near top and bottom of core

Objective 8

Figure 5-5 / TP 5-19



INSTRUCTOR GUIDE

KEY POINTS, AIDS, QUESTIONS/ANSWERS

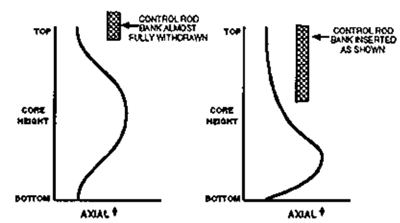
c. Anything affecting this flux distribution affects differential worth of rods

9. Movement of rods changes flux shape and values of differential rod worths

   a. The flux is depressed in region with rods inserted and is relatively higher in region without rods

   b. The flux distribution shifts as rods move into or out of core. Figure 5-6 shows flux shift from core midplane to core bottom as rods are inserted

Figure 5-6 / TP 5-20

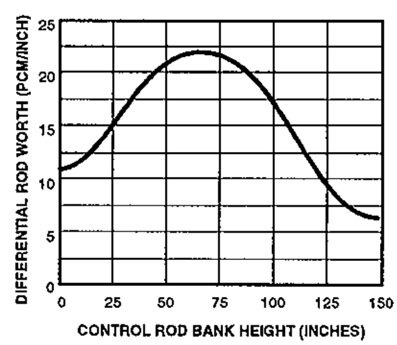


10. When control rods are near bottom, maximum flux shifts back to core midplane

    a. Because of this flux shift, highest differential rod worth occurs at rod height below core midplane

    b. A graph of differential rod worth versus rod height in typical reactor with banked control rods is given in Figure 5-7

Figure 5-7 / TP 5-21



    c. A rod bank is group of control rods which move together

52. CTMT PURGE-B.4 001

0-SO-30-3 contains the following precaution: "IF operating the containment purge system in MODE 5 or 6 WHILE the other unit is in MODES 1 through 4 THEN, an operator shall be available to stop the containment purge system in the event of an ABI."

Which ONE (1) of the following describes the basis for this precaution?

A. During containment purge system operation with the unit in MODE 5 or 6 its isolation function on a high rad condition is blocked and an operator is required to isolate the system.

B. The volume of air, that could pass from the unit being purged to the ABSCE via the containment purge fans and the open blast doors, would exceed the capacity and design basis of the EGTS.

✓C. The volume of air, that could pass from the unit being purged to the ABSCE via the containment purge fans and the open blast doors, would exceed the capacity and design basis of the ABGTS.

D. During MODE 5 or 6 the automatic Safety Injection function is blocked which inhibits the ABI signal. AND an operator is required to isolate the system.

a. Incorrect - Not a consideration in 0-SO-30-3
b. Incorrect - EGTS is not considered.
c. Correct - per reference material LER 50-327/88007 R5
d. Incorrect - blocking auto SI does not inhibit ABI signal, see 1,2-47W611-30-6.

K/A [CFR]: 2.3.9 [2.5/3.4] [43.4]

References: 0-SO-30-3 "Containment Purge Systen Operation"
LER 50-327/88007 R5
1,2-47W611-30-6

LP/Objectives: OPL271CONTPURGE B.7

History: Systems bank

Level: Memory

Comments: FHW 12/02 2.3.9

3.0 PRECAUTIONS AND LIMITATIONS (Continued)

D. The ice condenser doors must be monitored closely during all manipulations of the containment purge system. A slight imbalance in pressure between the upper and lower compartments will cause the doors to open. The Instructions must be followed exactly to minimize the probability of opening the ice doors.

E. During outage conditions, a flow imbalance can be created when a lower/upper compartment purge is inservice and changes are made to the upper and/or lower airlock doors (i.e. breaching or closing doors after purge is started). This situation may cause the Ice Condenser lower inlet doors to go open.

F. Operation of containment purge in Mode 5 or 6 with the equipment hatch closed and airlock doors breached may result in pressurization of the Reactor Building access rooms. This occurs due to thermal expansion of the colder outside air entering containment. Closing FCV-30-16 and FCV-30-17 and/or opening FCV-30-37 and FCV-30-40 in accordance with the applicable steps in this procedure will reduce the pressure in containment and reduce the risk to personnel entering and exiting the access rooms. This should also be used when lifting the head and moving fuel to reduce the spread of contamination.

G. When a containment ventilation isolation is initiated by a containment purge air exhaust radiation level signal, whether it is known to be spurious or not, this signal should be cleared from the circuit before resetting containment ventilation isolation.

H. <u>IF</u> operating the containment purge system in MODE 5 or 6 <u>WHILE</u> the other unit in MODES 1 through 4 <u>THEN</u>, an operator shall be available to stop the containment purge system in the event of an ABI. This action can be performed with minimum Tech Spec shift crew without impeding mitigation of the event. **[C.1]**

I. During MODE 6 operation with the wafer valve open, containment purge startup or shutdown may result in over flow of the SFP or reactor cavity due to pressure changes in the containment building.**[C.2]**

J. When air temperature entering the Auxiliary Building heating/cooling coils is $\leq 35°F$, flow must be maintained through the coils unless they are isolated and drained.

SOURCE NOTES
Page 1 of 1

REQUIREMENTS STATEMENT	SOURCE DOCUMENT	IMPLEMENTING STATEMENT
TVA will revise SOI-30.2 (SO-30-3) to include the appropriate CM that must be used when operating the containment purge system in modes 5 and 6 with the blast doors open while the other unit is in modes 1 through 4. Deleted blast doors open requirement due to DCN M01443.	LER 88007 R2 S53 880916 892	C.1
Overflow of the spent fuel pit and the reactor cavity has occurred at several plants due to pressure changes in either building during operations which change the configuration of the ventilation systems.	NER 91 0067001 INPO SER 91-001	C.2

1 ER File

8110 910531-839

TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

Jack L. Wilson
Vice President, Sequoyah Nuclear Plant

May 30, 1991

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - DOCKET NOS. 50-327 AND 50-328 - FACILITY OPERATING LICENSES DPR-77 AND 79 - LICENSEE EVENT REPORT (LER) 50-327/88007, REVISION 5

The enclosed LER has revised the long-term corrective actions to ensure that the auxiliary building gas treatment system can perform its design function during various modes of 2-unit operation.

As a result of a further review of the proposed modification to interlock Units 1 and 2 containment purge systems with the auxiliary building isolation signal as presented in Revision 4 to this LER, TVA has determined that the most feasible long-term solution would be the continued use of the compensatory measures currently in place. This decision considered the acceptability of maintaining the existing controls in the long-term, the performance in the last two outages utilizing the existing controls, the technical issues involved, and the cost and/or benefit of the alternatives.

This event was originally reported in accordance with 10 CFR 50.73, paragraph (a)(2)(i)(b), on February 23, 1988.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

JLWilson

J. L. Wilson

Enclosure
cc: See page 2

JLW:MAC:CHW:KRR

Enclosure

cc (Enclosure):

INPO Records Center
Institute of Nuclear Power Operations
1100 Circle 75 Parkway, Suite 1500
Atlanta, Georgia 30339

Mr. D. E. LaBarge, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

Mr. B. A. Wilson, Project Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

RIMS, MR 2F-C
R. J. Beecken, POB 2B-SQN
J. R. Bynum, LP 3B-C
W. R. Cobean, Jr., LP 3B-C
D. L. Conner, STC 2H-SQN
M. A. Cooper, OPS 4C-SQN
(Attn: J. S. Smith)
J. H. Garrity, FSB 1A-WBN
R. L. Lumpkin, Jr., SB 1C-SQN
R. W. Martin, OPS 4B-SQN
(Attn: T. J. Hollomon)
V. D. McAdams, SB 2B-SQN
T. J. McGrath, LP 3B-C
M. O. Medford, LP 3B-C
W. J. Museler, OSA 2A-BLN
D. A. Nauman, LP 3B-C
Nuclear Experience Review Files, OPS 4D-SQN
P. G. Trudel, DSE 1A-SQN
E. G. Wallace, LP 5B-C
O. J. Zeringue, PAB E-BFN

1442h

LICENSING TRANSMITTAL TO NRC
SUMMARY AND CONCURRENCE SHEET

DATE TRANSMITTED
TO MJR ____________

THE PURPOSE OF THIS CONCURRENCE SHEET IS TO ASSURE THE ACCURACY AND COMPLETENESS OF TVA SUBMITTALS TO THE NRC.

DATE ____________ DATE DUE NRC ____________ ACTION NO. ____________

SUBMITTAL PREPARED BY C. H. Whittemore FEES REQUIRED YES ____ NO X

PROJECT/DOCUMENT I.D. Sequoyah Nuclear Plant (SQN) - Licensee Event Report (LER) 50-327/88007, Revision 5

PURPOSE/SUMMARY To revise the long term corrective actions to ensure that the auxiliary building gas treatment system (ABGTS) can perform its design function during all modes of 2-unit operation.

RESPONDS TO N/A (RIMS NO.) COMPLETE RESPONSE YES X NO ___

PROBLEM OR DEFICIENCY DESCRIPTION Operability of ABGTS could not be assured because the auxiliary building secondary containment enclosure was not always maintained within the configuration set during technical specification testing used to verify ABGTS operability.

CORRECTIVE ACTION/COMMITMENT TVA has reviewed and evaluated long term alternatives and has determined that the most feasible long term solution to the problem is to continue to implement the measures currently in use.

INDEPENDENT REVIEW N/A DATE ____________

A concurrence signature reflects that the signatory has assured that the submittal is appropriate and consistent with TVA policy, applicable commitments are approved for implementation, and supporting documentation for submittal completeness and accuracy has been prepared.

CONCURRENCE

NAME	ORGANIZATION	SIGNATURE	DATE
R. J. Beecken	for SQN Plant Manager	W. R. Lagergren	5-28-91
PORC Chairman	W. R. Lagergren	W. R. Lagergren	5-28-91
M. A. Cooper	SQN Site Lic Mgr	M. A. Cooper	5/28/91

APPROVED James W Proffitt DATE 5/31/91
NLRA MANAGER

1442h

Sequoyah Site Licensing

Concurrence Sheet

DUE TO NTD ______________ DATE DUE NRC ______________

PROJECT/DOCUMENT I.D. Sequoyah Nuclear Plant (SQN) - Licensee Event Report (LER) 50-327/88007, Revision 5

CONCURRENCE

NAME	ORGANIZATION	SIGNATURE OR LETTER REFERENCE	DATE
C. H. Whittemore	SQN Licensing Engineer	[signature]	5/28/91
KRE J. W. Proffitt	SQN Compliance Lic Mgr	James W. Proffitt	5/23/91
J. S. Smith	SQN Site Lic	Jerry S. Smith	5/24/91
R. W. Martin	SQN Site Controller	NA - NO COMMITMENT CHW	5/28/91
J. K. Gates	SQN Technical Support Mgr	[signature]	5/23/91

NRC response or approval required? _____ Yes __X__ No

NOTE: This sheet should be removed by Corporate Licensing upon receipt.

1442h

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

Approved OMB No. 3150-0104
Expires 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Sequoyah Nuclear Plant, Unit 1

DOCKET NUMBER (2): 0|5|0|0|0|3|2|7

PAGE (3): 1 OF 0 7

TITLE (4) Opening of Unit 1 Containment Results in Containment Envelope Outside the Boundary Set for Surveillance Testing of Auxiliary Building Gas Treatment System

EVENT DAY (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 1	2 4	8 8	8 8	0 0 7	0 5	0 5	3 0	9 1	Sequoyah, Unit 2	0|5|0|0|0|3|2|8
										0|5|0|0|0| | |

OPERATING MODE (9): 1

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following)(11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ OTHER (Specify in Abstract below and in Text, NRC Form 366A)
__ 20.405(a)(1)(iii)	XX 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: C. H. Whittemore, Compliance Licensing Engineer

TELEPHONE NUMBER: AREA CODE 6 1 5 | 8 4 3 - 7 2 1 ...

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	V I	A H U	E 3 2 2	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

__ YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

This LER has been revised to describe the results of an evaluation to ensure that the auxiliary building gas treatment system (ABGTS) remains operable during 2-unit operation. On January 24, 1988, with Units 1 and 2 in Mode 5 (cold shutdown), it was discovered that the auxiliary building secondary containment enclosure was not being maintained within the configuration set during the technical specification (TS) surveillance testing used to verify ABGTS operability. On August 24, 1988, with Unit 1 in Mode 5 and Unit 2 in Mode 1, it was determined that the Unit 1 containment purge system was in operation without the required compensatory measures being properly documented. These conditions were caused by (1) the lack of adequate controls to ensure the ABSCE boundary was maintained within the condition set by surveillance testing, (2) an inappropriate design assumption made during plant construction on how ABSCE breaches would be controlled, and (3) an incomplete compensatory measures (CM) program. Corrective Actions included closing the Unit 1 blast door and tagging the Unit 1 containment purge system out of service, revising the procedure governing ABSCE breaches, and upgrading the CM program. Following subsequent leak testing of the Unit 1 annulus, the Unit 1 blast door was reopened. As long-term corrective action, TVA has instituted administrative controls through the CM program to ensure TS requirements are satisfied under the subject conditions.

This LER has been revised to describe the results of the evaluation considering alternatives to ensure that the ABGTS can perform its design function during various modes of 2-unit operation.

DESCRIPTION OF CONDITION

On January 24, 1988, with units 1 and 2 in mode 5 (0 percent power, 4 psig, 121 degrees F and 0 percent power, 310 psig, 118 degrees F, respectively), a potential deficiency in the Auxiliary Building secondary containment enclosure (ABSCE) (EIIS Code WF) was discovered during a tour of the refueling area and subsequent discussions with test personnel. The plant configuration used when testing the ABSCE in accordance with Technical Specification (TS) Surveillance Requirement (SR) 4.7.8.d.3 was not consistent with allowable plant configurations during various modes of two unit operation. As a result, operability of the ABGTS could not be assured, and Condition Adverse to Quality Report (CAQR) SQP 880090 was issued.

The ABGTS and the ABSCE are common to units 1 and 2, which share a common Auxiliary Building (EIIS Code NF). Both trains of the ABGTS are required to be operable before either unit can enter mode 4 from a mode 5 condition. The ABGTS maintains negative pressure in the ABSCE and filters the ABSCE air before it is released to the environment. One ABGTS train is required to be operable for unrestricted fuel handling operations while irradiated fuel is in the spent fuel pool (although the ABGTS is not required to maintain a negative pressure in the ABSCE during plant operations in modes 5 and 6).

TS SR 4.7.8.d.3 requires verification that the ABGTS can maintain the spent fuel storage area and the engineered safety feature (ESF) pump rooms within the ABSCE at a pressure equal to or more negative than minus 1/4-inch water gage (wg) while maintaining a vacuum relief flow rate greater than 2000 cubic feet per minute (cfm) and a total system flow rate of 9000 cfm $\pm$ 10 percent. This SR is satisfied by the performance of Surveillance Instruction (SI)-149, "Auxiliary Building Gas Treatment System Vacuum Test." Past performances of SI-149 had both the unit 1 and unit 2 blast doors (refueling floor to containment annulus doors on the 734 feet elevation) in the Reactor Building shield walls closed, and containment purge on both units shut down.

During plant operation in modes 5 or 6, however, it is normal for that unit to have its blast door and/or equipment hatch open. Opening the blast door increases the ABSCE, boundary by the addition of the annulus. If the equipment hatch or personnel access doors are also open, the ABSCE boundary is increased further by the addition of the primary containment. The increased boundary causes additional leakage into the ABSCE that was not accounted for during the previous performances of SI-149.

Thus, if one unit is in mode 5 or 6 with the blast door/equipment hatch open, and the opposite unit is in modes 1, 2, 3, or 4 (i.e., an operational mode that requires the ABGTS to be operable), the actual plant configuration would not be the same as the configuration that was tested during the performance of SI-149.

DESCRIPTION OF CONDITION (continued)

A second concern that has been identified as potentially affecting the performance of the ABGTS during an accident relates to the operation of the containment purge system on a unit with the blast door and equipment hatch open. The containment purge system, when it is operating, provides a large amount of air into the Reactor Building (EIIS Code NH). Air contributed from the containment purge system was not accounted for during the performance of SI-149, and its operational status was not being controlled with the opening of the blast doors and the equipment hatch. Thus, there was no assurance that TS SR 4.7.8.d.3 could be satisfied if the blast door and equipment hatch were open, and the containment purge system for that unit was in operation.

In order to allow unit 2 to enter mode 4 (which occurred on February 6, 1988), TVA administratively prohibited the operation of the unit 1 containment purge system whenever the equipment hatch and blast door were open by implementing the provisions of temporary alteration change form (TACF) 1-88-02-030. This TACF, which was approved on January 28, 1988, placed hold order 1-88-240 on the unit 1 containment purge fans, thereby preventing their operation. In addition to implementing the TACF, TVA performed SI-264, "EGTS Annulus Vacuum Draw Down Test," to measure the leakage into the unit 1 annulus. This leakage was then conservatively added to the previously measured ABSCE leakage to verify that the ABGTS could perform its intended function with the blast door open.

Following further investigation into this event, it was determined that there was a need to demonstrate that operation of the containment purge system in a unit that had established containment integrity would not have an adverse effect on the ability of the ABGTS to draw down the ABSCE to minus 1/4-inch wg within the 1-minute time interval specified in the Final Safety Analysis Report (FSAR). That is, even with containment integrity established, it was postulated that the containment purge system duct work in the Auxiliary Building could leak and prevent the ABGTS from performing its design function.

To verify the integrity of the purge system duct work, TVA performed smoke tests and visual inspections of the subject duct work in accordance with SI-506.7, "Containment Purge Air Exhaust Filter Train Test." However, performance of this test required operation of the containment purge system which had been tagged out of service by TACF 1-88-02-030. In order to operate the purge system, a compensatory measure was approved to allow operation of the system as long as operator action was taken within four minutes of an Auxiliary Building Isolation (ABI) signal (EIIS Code JE) to shutdown the system. Temporary Instruction Change Form (ICF) 88-890 and permanent ICF 88-0977 were subsequently approved to incorporate this compensatory measure into SI-506.7.

A similar (but temporary) ICF was written against SOI-30.2, "Containment Purge System Operation, to allow a one-time operation of the purge system to reduce an unexpected increase in the containment airborne radiation level on July 25, 1988.

DESCRIPTION OF CONDITION (continued)

On August 24, 1988, a revision to TACF 1-88-02-030 was presented to the Shift Operations Supervisor (SOS) for implementation. This revision changed the tagging boundary from both trains of containment purge isolation valves to only one train of valves to allow SI-26, "Loss of Offsite Power with Safety Injection - D/G Containment Isolation Test," to be performed.

Upon receiving the revision to TACF 1-88-02-030, the SOS realized that the unit 1 containment purge system was being run at that time (in accordance with SOI-30.2) to reduce the temperature inside the unit 1 containment. However, since the ICF to SOI-30.2 had expired, the subject SOI did not have the appropriate compensatory measure for purge system operation. The SOS immediately suspended purge system operation, reissued the hold order on the system, and requested an investigation be initiated. This investigation revealed that, although most operators were aware of the compensatory measures necessary for operating the unit 1 purge system, these measures had not been adequately documented in SOI-30.2, nor were they formally communicated to Operations personnel. Thus, there was no assurance that plant operators would have shut down the unit 1 containment purge system following an ABI signal, and as a result, there was no assurance that the ABGTS would have been able to perform its design function.

CAUSE OF CONDITION

The immediate cause of this condition was the failure to ensure the ABSCE configuration was maintained in the same configuration that was set during surveillance testing of the ABGTS in accordance with SI-149. TS 3.6.1.1 requires primary containment integrity only for a unit that is in modes 1 through 4. TS 3.7.8 requires the ABGTS to be operable whenever either unit is in modes 1 through 4. However, operability of the ABGTS was verified only with the blast doors closed. Breaches of the ABSCE are controlled by Technical Instruction (TI)-77, "Breaching the Shield Building, ABSCE, or Control Room Boundaries." However, this TI did not properly evaluate the condition when (1) the Shield Building boundary becomes part of the ABSCE (through an open blast door), (2) the primary containment becomes part of the ABSCE (if the equipment hatch and blast door are open), or (3) the containment purge system is in operation.

The root cause of this event was improper design assumptions that were made during the period of plant construction to address breaches in the ABSCE. The need for an interim ABSCE was recognized (and provided) during the time one unit was in operation and the other unit was still under construction. At that time, it was also recognized that upon completion of both units, there would be times when the need to breach the ABSCE would exist. However, it was believed at that time that most ABSCE breaches would be of short duration and could be justified based on the low probability of an accident during that time.

CAUSE OF CONDITION (continued)

It was expected that long duration breaches for major modifications would be compensated for by establishing an interim ABSCE similar to that established during construction. However, this design philosophy was not documented at that time because no formal procedure existed that required this type of documentation.

Running the containment purge system without formal compensatory measures established was caused by an incomplete compensatory measures (CM) program instituted by Administrative Instruction (AI)-49, "Control and Tracking of Compensatory Measures." A review of the compensatory measures program has shown that, although the program appears to be appropriate for tracking and evaluating the effectiveness of CMs once they are identified, there are no specific guidelines that require CMs to be considered.

Specifically, a review of implementing documents for (1) performing safety evaluations, (2) performing procedure changes, and (3) performing temporary facility changes (TACFs) failed to identify any requirements for evaluating these changes for necessary CMs.

Further review of the CM program revealed that, once a CM has been deemed appropriate, there is only one step in AI-49 which requires the CM program manager to ensure that the implementing organization is aware of the CM. Although this step is certainly appropriate, there was no clear method for it to be accomplished. Specifically, administrative measures to disseminate information to shift operating crews concerning CMs were not standardized, and consequently, were inadequate. In addition, there was no administrative control in place that required existing CM information to be passed on during shift turnover.

ANALYSIS OF CONDITION

This condition was originally reported under 10 CFR 50.73, paragraph a.2.i.b, as a condition prohibited by TS.

TS SR 4.7.8.d.3 is performed as a partial verification that the ABGTS is operable and capable of performing its design function. Since the actual plant configuration was nonconservatively different from the configuration used when testing the ABGTS in accordance with TS SR 4.7.8.d.3, there was no assurance that the ABGTS would have satisfied its design function.

The condition as discovered, however, was not considered to have had a significant safety consequence to the health and safety of the public because units 1 and 2 were in cold shutdown, and the ABGTS was not required to satisfy TS SR 4.7.8.d.3 during plant operation in modes 5 or 6. In addition, no fuel handling operations were in progress in the spent fuel pool area.

However, there have been occasions when a blast door has been open while the opposite unit was not in modes 5 or 6.

CORRECTIVE ACTIONS

If a LOCA had occurred while a unit was in modes 1, 2, 3, or 4, fission products could have been released to the ABSCE. If the fission products were released to the ABSCE while the blast door and equipment hatch were open (and that unit was operating its containment purge system), there would be no assurance that all radioactive materials leaking from the ESF equipment or from primary containment into the ABSCE would be filtered by the ABGTS filters before reaching the environment. This postulated event would then be outside the assumptions made in the offsite dose calculations for accident analysis. However, the ABGTS filters were available for filtration of air released from the ABSCE, and containment exhaust filters are used to filter air released from the primary containment when the containment purge system is operating.

As described previously, the short-term corrective action consisted of closing the unit 1 blast door and tagging the unit 1 containment purge system out of service before unit 2 entered operational mode 4 (which occurred on February 6, 1988). To allow opening the blast door of a unit in modes 5 or 6 while the opposite unit is in modes 1, 2, 3, or 4, TI-77 was changed in accordance with ICF 88-0191. Incorporation of the provisions of this ICF establishes administrative controls to ensure that the requirements of TS SR 4.7.8.d.3 are satisfied when one unit's blast door and/or equipment hatch is open and the other unit is in modes 1, 2, 3, or 4. To account for the additional leakage when the primary containment and annulus become part of the ABSCE, the maximum expected leakage of this area was calculated and subtracted from the tolerance by which the ABGTS flowrate required to satisfy TS SR 4.7.8.d.3 was exceeded. The remaining tolerance was then used to determine the cumulative area that can be breached and still satisfy TS SR 4.7.8.d.3.

The maximum expected leakage was based on the FSAR value of 500 cfm. Test data from the most recent performance of SI-264 verified that the leakage into the annulus was well within the 500 cfm limit. In addition, the majority of this leakage is from the Auxiliary Building which would not be classified as ABSCE leakage when a blast door is open.

To ensure adequate consideration is given to establishing necessary CMs, TVA has reviewed appropriate plant procedures (e.g., AI-4, "Preparation, Review, Approval, and the Use of Site Procedures/Instructions;" AI-9, "Control of Temporary Alterations Order;" AI-19, Part VI: "Modifications; Permanent Design Change Control Program;" and SQA-119, "Safety Evaluations") to determine if the subject procedures should be revised to require personnel using these procedures to determine if compensatory measures are involved. To ensure that the Auxiliary Building gas treatment system can perform its design function during various modes of two unit operations, TVA has enhanced SOI-30.2 Abnormal Operating Instruction (AOI)-6, "Small Reactor Coolant System Leak," AOI-31, "Abnormal Release of Radioactive Materials," and Emergency Operating Instruction E-0 "Reactor Trip or Safety Injection," such that a TACF will not be required to continuously remove the containment purge system from service during the unit 2 cycle

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah Nuclear Plant Unit 1	0 5 0 0 0 3 2 8	9 1	0 0 4	0 5	0 7 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

CORRECTIVE ACTIONS (continued)

refueling outage. These enhancements ensure that adequate CMs will be taken to ensure that the ABGTS will perform its designed function if required and allows for operation of the containment purge system when required. In addition to the above described procedure changes, TVA has established requirements for a technical review of all active CMs on a periodic basis. This review will verify that all the assumptions that were originally used to justify a particular CM remain valid.

Since Operations personnel are responsible for implementing almost all CMs, TVA has established a CM log book in the main control room that contains all active CMs. In addition, AI-5, "Shift Relief and Turnover," has been revised to require appropriate Operations shift personnel to review the active CMs before they assume shift.

To prevent recurrence of this type of event in the future, TVA has implemented design control procedures that requires documentation of quality information and communication between design organizations and/or Operations groups on site. Specifically, Nuclear Engineering Procedure (NEP)-5.3, "External Interface Control," establishes controls for interactions between organizations outside Nuclear Engineering (NE) to ensure the appropriate transfer of information necessary to accomplish engineering, design and related services for TVA. In addition, NEP-5.2, "Review," ensures that reviews done within NE include an appropriate Operation and Maintenance data review.

ADDITIONAL INFORMATION

There has been one previous occurrence reported in the ABGTS failing to meet TS SR because of improper ABSCE boundary control - SQRO-50-327/84053.

53. CTMT-B.11 001

Which ONE ~~(1)~~ of the features stated below is designed to provide more efficient operation of the Ice Condenser and Containment Spray for heat removal after blowdown resulting from a LOCA?

A. The Divider Barrier Seal between upper and lower containment will open to enhance mixing of the two environments.

B. Lower containment Ventilation Fans are designed to mix the air and provide additional cooling during a loss-of-coolant accident.

✓C. The containment Air return fans provide flow to return air from upper containment to lower containment during a LOCA.

D. Pressure-operated Ice-Condenser doors will open to allow upper containment air to flow to lower containment.

A. Incorrect, the divider barrier seal must be operable to enhance flow through the ice condenser.
B. Incorrect per TS bases, only considered for a non LOCA accident.
C. Correct per TS bases.
D. Incorrect the doors were opened for flow from the lower to upper.

Student must comprehend containment conditions, equipment status during and after blowdown.

K/A[CFR]:	025 K5.02 [2.6/2.8] [41.5]
Reference:	TS bases for 3.6.5.6 and 3.6.5.9
LP/Objective:	OPL271C024, b.11
History:	Old Bank Number PL-0227. Modified distractor A to ensure one correct answer, FHW.
Level:	~~Comprehension.~~ memory
Comments:	FHW 12/02 025 K5.02

event that observed sublimation rates are equal to or lower than design predictions after three years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the ice bed temperature monitoring system ensures that the capability is available for monitoring the ice temperature. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors ensures that these doors will open because of the differential pressure between upper and lower containment resulting from the blowdown of reactor coolant during a LOCA and that the blowdown will be diverted through the ice condenser bays for heat removal and thus containment pressure control. The requirement that the doors be maintained closed during normal operation ensures that excessive sublimation of the ice will not occur because of warm air intrusion from the lower containment. BR

If an ice condenser inlet door is physically restrained from opening, the system function is degraded, and immediate action must be taken to restore the opening capability of the inlet door. Being physically restrained from opening is defined as those conditions in which an inlet door is physically blocked from opening by installation of a blocking device or by an obstruction from temporary or permanently installed equipment or is otherwise inhibited from opening such as may result from ice, frost, debris, or increased inlet door opening torque beyond the values specified in Surveillance Requirement 4.6.5.3.1. R165

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

The OPERABILITY of the inlet door position monitoring system ensures that the capability is available for monitoring the individual inlet door position. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.5 DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

The requirements for the divider barrier personnel access doors and equipment hatches being closed and OPERABLE ensure that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of the steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.5.6 CONTAINMENT AIR RETURN FANS

The OPERABILITY of the containment air return fans ensures that following a LOCA 1) the containment atmosphere is circulated for cooling by the spray system and 2) the accumulation of hydrogen in localized portions of the containment structure is minimized.

3/4.6.5.7 and 3/4.6.5.8 FLOOR AND REFUELING CANAL DRAINS

The OPERABILITY of the ice condenser floor and refueling canal drains ensures that following a LOCA, the water from the melted ice and containment spray system has access for drainage back to the containment lower compartment and subsequently to the sump. This condition ensures the availability of the water for long term cooling of the reactor during the post accident phase.

3/4.6.5.9 DIVIDER BARRIER SEAL

The requirement for the divider barrier seal to be OPERABLE ensures that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.6 VACUUM RELIEF LINES R201

The OPERABILITY of three primary containment vacuum relief lines ensures that the containment internal pressure does not become more negative than 0.1 psid. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of 0.5 psid. A vacuum relief line consists of a self-actuating vacuum relief valve, a pneumatically operated isolation valve, associated piping, and instrumentation and controls.

54. CTMT-B.11 004

During outages the CRDM motor power supply may be temporarily realigned to supply receptacle power in the lower containment.

Which ONE ~~(1)~~ of the following describes how this condition is controlled?

With the motor breaker racked out:

A. the motor leads are lifted and re-landed on the receptacle power pack. A TACF is placed on the breaker to identify this temporary condition.

B. the motor leads are lifted and re-landed on the receptacle power pack. A Hold Order is placed on the breaker to identify this temporary condition.

C. a transfer switch is aligned to supply power to the receptacle power pack. A TACF is placed on the breaker to identify the breaker is under receptacle load.

✓D. a transfer switch is aligned to supply power to the receptacle power pack. A Caution Order is placed on the MCR handswitch to identify the breaker is under receptacle load.

A. A transfer switch is used so the leads will not have to be lifted and re landed.
B. A transfer switch is used so the leads will not have to be lifted and re landed.
C. A transfer switch is used , however a CO is used rather than a TACF.
D. Correct per reference.

K/A {CFR}:	SYS 022 2.2.11	[2.5/3.4]	{41.10, 43.3}
	022 K2.01	[3.0/3.1]	{41.7}
	2.2.11	[2.5/3.4]	[41.10 43.3]

References: 0-SO-30-6 R17 section 8.2 [9]

LP/Objectives: OPL271CONTCOOLING B.6

History: System bank

Level: Memory

Comments: LP-5/2000. CV 001; FHW 12/02 2.2.11, modified selected correct answer to conform with 0-SO-30-6 as to "handswitch" not "breaker." This is a unit difference question.

Is this a TC or just a realignment

No TACF Required

Unit 1 Date ____________

8.2 Transfer of CRDM Cooling Fan Motor Power Supply from Fan to 480v Receptacle Inside Lower Containment During Outages for Unit 1 ONLY. (Continued)

NOTE The handswitches for the CRDM cooling fans are spring returned to mid-position.

[8] **PLACE** applicable handswitch for CRDM Cooling Fan in **START** position.

A. **[1-HS-30-83A]** 1A CRDM Cooling Fan ______

B. **[1-HS-30-92A]** 1B CRDM Cooling Fan ______

C. **[1-HS-30-88A]** 1C CRDM Cooling Fan ______

D. **[1-HS-30-80A]** 1D CRDM Cooling Fan ______

[9] **PLACE** Caution Order on applicable handswitch that warns NOT to manipulate handswitch due to breaker being under receptacle load.

A. **[1-HS-30-83A]** 1A CRDM Cooling Fan ______

B. **[1-HS-30-92A]** 1B CRDM Cooling Fan ______

C. **[1-HS-30-88A]** 1C CRDM Cooling Fan ______

D. **[1-HS-30-80A]** 1D CRDM Cooling Fan ______

END OF TEXT

55. CTMT-B.5 004

This question has reference material attached.

Unit 1 is operating at 100% power. Monthly containment entry inspections are in progress. Which ONE (1) of the following would result in a loss of Primary Containment Integrity?

A. An AUO makes an entry into the Unit 1 Annulus and leaves the El 690 Annulus Entry Door unlatched.

B. Primary Containment internal pressure is discovered to be at .75 psig relative to Annulus pressure.

✓C. The DI water system was discovered unisolated and unattended to lower containment service outlets.

D. The Upper Containment Personnel Airlock door operating mechanism is broken and the door is locked closed.

A. Incorrect, the annulus door is not a containment air lock door.
B. Incorrect, outside the limits of LCO 3.6.1.4.
C. Correct per LCO 3.6.1.1.
D. Incorrect, actions of LCO 3.6.1.3 have been met.

K/A: 103 K3.02 [3.0 / 4.2]
069 AA1.03 [2.8/3.0] {41.7}

Reference: Tech Spec LCO 3.6.1.1
Tech Spec LCO 3.6.1.4
Tech Spec LCO 3.6.3
1-SI-OPS-088-014.0 R13

Objective: OPL271C083, B.3

History: Systems Bank

Level: Comprehension

Comments: **Provide tech specs 3.6.3, 3.6.1.1, 3.6.1.4** FHW 12/02 069 AA1.03

NOTE **Prior to closing** the following valves, US/SRO or WCC/SRO approval is required to verify that closure will not affect in-progress work.

Approval for closure (check appropriate valve(s)): ☐ 1-VLV-59-522

☐ 1-VLV-33-740

____

US/SRO or WCC/SRO (circle one)

690 PENETRATION ROOM						
EQUIPMENT ID	REQUIRED POSITION	LOCATION	PENETRATION	PRINT REFERENCE	INITIALS	
1-59-522 DI water to cntmt	LOCKED CLOSED	Behind RCP seal flow indicators south wall	X-77	47W856-1	1st	IV
1-33-740 Service Air to Cntmt	LOCKED CLOSED	Behind RCP seal flow indicators south wall	X-76	47W846-2	1st	IV

Date____________

6.0 PERFORMANCE

NOTE 1 Independent verification is only required for initial performance and following valve, blind flange, or plug manipulation. Otherwise, the independent verification may be N/A'd.

NOTE 2 Containment Boundary Tags (App H of 0-GO-15) may be picked up during the performance of this instruction provided the RCS is no longer in Reduced inventory/Midloop condition.

[1] **ENSURE** that Prerequisites in Section 4.0 are met. ________

NOTE 1 Appendixes A & B are to be completed by Operations. Appendixes C and D are to be completed by Instrument Maintenance Group.

NOTE 2 Refer to Technical Instruction 0-TI-OPS-088-001.0 as needed.

NOTE 3 The following prints may be useful in identifying and locating the penetrations: 47W470-2, 47W253-2, 48N405, 48N406.

NOTE 4 Appendixes not required for this performance may be N/A'd.

[2] **IF** Appendix A is scheduled to be performed, **THEN COMPLETE** Appendix A as follows:

[a] **LIST** any open Tech Spec 3.6.3 LCO action items and associated method of isolation on Table B-1 of Appendix A. ________

[b] **VERIFY** the method of isolation for each item listed on Table B-1. ________

[c] **VERIFY** all portions of Appendix A are completed. ________

[3] **IF** Appendix C is scheduled to be performed, **THEN COMPLETE** Appendix C. ________

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. Deleted. R258

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3. R134

c. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program. R221

R258

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and 0.3 psig relative to the annulus pressure.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to within the limits at least once per 12 hours.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.* R207

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more penetration flow paths with one containment isolation valve inoperable; except for containment vacuum relief isolation valves(s), isolate each affected penetration within 4 hours by use of at least one closed deactivated automatic valve, closed manual valve, blind flange, or check valve## with flow through the valve secured; and, verify# the affected penetration flow path is isolated once per 31 days for isolation devices outside containment, and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment. R258

b. With one or more penetration flow paths with two containment isolation valves inoperable; except for containment vacuum relief isolation valves(s), isolate each affected penetration within 1 hour by use of at least one closed deactivated automatic valve, closed manual valve, or blind flange and verify# the affected penetration flow path is isolated once per 31 days.

c. With one or more containment vacuum relief isolation valve(s) inoperable, the valve(s) must be returned to OPERABLE status within 72 hours.

d. With any of the above ACTIONS not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Deleted R207

---

*1. Penetration flow path(s) may be unisolated intermittently under administrative controls.

2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

#3. Isolation devices in high radiation areas may be verified by use of administrative means.

#4. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. R258

##5. A check valve with flow through the valve secured is only applicable to penetration flow paths with two containment isolation valves.

4.6.3.2 Each automatic containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by: R258

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Ventilation isolation test signal, each Containment Ventilation Isolation valve actuates to its isolation position.

d. Verifying that on a high containment pressure isolation test signal, each Containment Vacuum Relief Valve actuates to its isolation position. R85

e. Verifying that on a Safety Injection test signal that the Normal Charging Isolation valve actuates to its isolation position. R105

4.6.3.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5. R207

4.6.3.4 At least once per 31 days, verify that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control.

R258

*Except valves, blind flanges and deactivated automatic valves which are located inside the annulus or containment or the main steam valve vaults and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

64. ECA-1.2-B.1 002

Given the following plant conditions:

- Unit 2 is operating at 100% power when a LOCA ~~outside containment is recognized.~~ occurs cnmt pressure 0.1 psid and steady
- The operators initiated safety injection.
- The crew has determined that SI cannot be terminated.
- SI has been reset.
- RCS pressure is 500 psig and dropping slowly.

Which ONE ~~(1)~~ of the following best describes the procedure methodology to mitigate this condition?

A. If the LOCA can **NOT** be isolated then transition to E-1.

✓B. If the LOCA can **NOT** be isolated then transition to the Loss of RHR Sump Recirculation procedure.

C. When the RWST level is <27%, the operator should transfer ECCS pumps to the containment sump.

D. When the RWST level is <8%, the operator should transfer the ~~containment spray~~ ECCS pumps pumps to the containment sump. ID

A. Incorrect - ECA-1.2 will not transition you to E-1.
B. Correct - per EPM-3 "Basis Document"
C. Incorrect - ECA-1.2 makes NO provisions for this evolution.
D. Incorrect - ECA-1.2 makes NO provisions for this evolution.

K/A CFR:	W/E04 2.4.4	[4.0/4.3]	{41.10, 43.2}
	E04 EK2.2	[3.8/4.0]	{41.7}

References: ECA-1.2

LP/Objectives: OPL271C418 B.1

History: Bank

Level: Memory

Comments: FHW 12/02 E04 EK2.2

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**DETERMINE** if LOCA isolated:	
	a. **CHECK** RCS pressure RISING.	a. **GO TO** ECA-1.1, Loss of RHR Sump Recirculation.
	b. **GO TO** E-1, Loss of Reactor or Secondary Coolant.	

END

67. ECCS-B.2 002

Which ONE ~~(1)~~ of the following ~~best describes~~ is a safety injection initiation signal?

A. High containment pressure of 2.81 psid on 2/4 pressure transmitters.

B. High containment pressure of 2.81 psid on 2/3 pressure transmitters with P-11 blocked.

C. High containment pressure of 1.54 psid on 2/4 pressure transmitters with P-11 blocked.

✓D. High containment pressure of 1.54 psid on 2/3 pressure transmitters.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A[CFR]: 00013 K1.01 (4.2-4.4) {41.2 -9}
E14 EK1.3 [3.3/3.6] {41.8 41.10}

Reference: TI-28

History: Systems bank old Bank Number PL-1468

LP/Objective: OPL271RPS b.10

Level: Memory

Comments: FHW 12/02 E14 EK1.3

E/S 2.1 3.4/3.7

missing remedial actions part of K/A

Signal	Setpoint	Logic	Block/Permissive

SAFETY INJECTION SIGNALS (SIS)

1. Containment Press Hi	1.54 psid	2/3 PTs	None
2. Pressurizer Press. Low	1870 psig	2/3 PTs	Manual Below P-11
3. Steamline Press Low	600 psig	2/3 PTs on 1/4 loops	Manual Below P-11
4. Manual	N/A	1/2 HS's	

SI RESET - AFTER SI INITIATION, MUST WAIT FOR 60 SECOND TIMER TO RESET. THEN THE SI RESET PB FOR EACH TRAIN MUST BE ACTUATED. THIS WILL BLOCK ANY AUTOMATIC SI ACTUATION SIGNAL BUT MANUAL SI IS NOT BLOCKED. TO REMOVE AUTO SI BLOCK, THE RX TRIP BREAKERS MUST BE CYCLE TO REMOVE THE P-4 SEAL-IN SIGNAL.

CONTAINMENT ISOLATION SIGNALS (CIS) Phase A

1. SIS	Any Signal		
2. Manual	Phase A / CVI HS	1/2	

CONTAINMENT ISOLATION SIGNALS (CIS) Phase B

1. Containment Press Hi-Hi	2.81 psid	2/4	
2. Manual	Phase B Handswitch	2/2	

CONTAINMENT VENT ISOLATION SIGNALS (CVI)

1. RM-90-130 & 131	High Rad Signal	1/2	
2. SIS	Any Signal		
3. Manual Phase B	Phase B Handswitch	2/2	
4. Manual Phase A	Phase A Handswitch	1/2	

CONTAINMENT SPRAY ACTUATION SIGNALS

1. Containment Press Hi-Hi	2.81 psid	2/4	
2. Manual	Phase B Handswitch	2/2	

MAIN STEAMLINE ISOLATION SIGNALS

1. Containment Press Hi-Hi	2.81 psid	2/4 PTs	
2. Steamline Press Low	600 psig	2/3 PTs on 1/4 loops	Manual Below P-11
3. Steamline Press Negative Rate	100 psig decreasing in a 50 second time constant	2/3 PTs on 1/4 loops	*Enabled only when Steamline Press SI signal blocked.*

FEEDWATER ISOLATION SIGNALS

1. S/G Level Hi-Hi	81% (P-14)	2/3 LTs on any S/G	
2. Rx Trip (P-4) with Lo T_{ave}	Rx Trip Bkrs Open 550 °F	2/4 loops	
3. SIS	Any signal		

68. ECCS-B.3 002

The unit is experiencing a large break LOCA. Which ONE ~~(1)~~ of the following best describes the order of ECCS component injection?

~~Highest pressure to lowest pressure.~~ SD

A. CCPs, SI Pumps, RHR Pumps, Cold Leg Accumulators.

✓B. CCPs, SI Pumps, Cold Leg Accumulators, RHR Pumps.

C. SI Pumps, CCPs, Cold Leg Accumulators, RHR Pumps.

D. SI Pumps, CCPs, RHR Pumps, Cold Leg Accumulators.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

Student must comprehend the order of injection based on the pump/accumlator discharge head.
K/A [CFR]: 006 A1.07 [3.3/3.6] [41.5]
006 K1.03 (4.2 - 4.3)
006 K5.06 (3.5 - 3.9)
006 A3.01 (4.0 - 3.9)
006 A3.02 (4.1 - 4.1)

Reference: ECCS System description section 3.

LP/Objective: OPL271ECCS B.13

History: Systems bank Old Bank Number PL-1530

Level: Memory

Comments: FHW 12/02 006 A1.07

Emergency Operations, (Continued)

ECCS process
Obj 10

The table below describes the ECCS process during a Large Break Loss of Coolant Accident (LBLOCA) in the RCS.

Pressurizer Pressure	Action
< 1870 psig	• A Safety Injection occurs. ECCS performs the following automatic actions: ➤ Both CCPs start (one CCP is normally running with RCS level at the RCP seals). ➤ Both SIPs start. ➤ Both RHR Pumps start. ➤ LCV-63-135, 136 open (CCP RWST suction). ➤ LCV-62-132, 133 close (CCP VCT suction). ➤ FCV-63-39, 40 open (CCP cold leg discharge). ➤ FCV-63-25, 26 open (CCP cold leg discharge). ➤ FCV-62-90 and 62-91 close. (Regenerative Heat Exchanger valves). ➤ FCV-63-118, 98, 80, 67 receive an open signal(should already be open) (Accumulator discharge). • The CCPs inject water into the RCS Cold Legs, since RCS pressure is less than the shutoff head of the CCPs (2250 psig). • The SIPs flow recirculates back to the RWST through valve FCV-63-3, since the shutoff head of the SIPs is 1500 psig. • The RHR pumps flow recirculates through valve FCV-74-12, 24, since the shutoff head of the RHR pumps is 180 psig.
< 1500 psig	The SIPs begin to inject water into the RCS Cold Legs.
< 640 psig	The Accumulators begin to inject water into the RCS Cold Legs, since the Accumulators nitrogen gas pressure is now greater than RCS pressure.
< 180 psig	The RHR pumps begin to inject into the RCS Cold Legs.

Continued on next page

70. ES-0.0-B.3 001

Given the following plant conditions:

- Reactor trip and SI have occurred 10 minutes ago
- RCS T_{cold} 535°F
- RCS Pressure 1810 psig
- Pressurizer level 15%
- S/G NR Levels 20% each
- S/G #1 Pressure 875 psig
- Other S/G Pressures 890 psig
- AFW Flow is 150 gpm to each S/G

Which ONE ~~(1)~~ of the following identifies the type of accident that is in progress?

✓A. Small break LOCA

B. Steam line break

C. Feed line break

D. S/G tube rupture

A. Correct for initial conditions.
B. Incorrect since all S/G pressure are approx. the same after 10 minutes.
C. Incorrect since all feed flows are the same.
D. Incorrect since all S/G levels are approx. 20% after 10 minutes.

K/A: 000009A202 [3.5/3.8]
E01 EA1.1 [3.7/3.7] {41.7}

Reference: ES-0.0, pages 4 & 5

Objective: OPL271C380, b.3

History: Procedure Bank

Level: Comprehension

Comments: FHW 12/02 E01 EA1.1

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**DETERMINE** procedure applicability: • **CHECK** any SI pump RUNNING. • **CHECK** CCPIT flow INDICATED. • **CHECK** E-0, Reactor Trip or Safety Injection, previously COMPLETED.	**IF** SI required, **THEN** **PERFORM** the following: a. **ACTUATE** SI. b. **GO TO** E-0, Reactor Trip or Safety Injection. **IF** SI **NOT** required, **THEN** **RETURN TO** procedure and step in effect.
2.	**CHECK** S/G secondary pressure boundary integrity: • Any S/G pressure stable or rising.	**IF** controlled cooldown in progress, **THEN** **GO TO** Step 3. **IF** controlled cooldown **NOT** in progress, **THEN** **PERFORM** one of the following: • **IF** main steamlines **NOT** isolated, **THEN** **GO TO** E-2, Faulted Steam Generator Isolation. OR • **IF** main steamlines isolated, **THEN** **GO TO** ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**CHECK** S/G secondary pressure boundary integrity: • S/G pressures controlled or rising • S/G pressures greater than 140 psig.	**VERIFY** all Faulted S/G(s) ISOLATED: • MSIVs and bypasses CLOSED • AFW ISOLATED • MFW ISOLATED • Atmospheric relief CLOSED • S/G blowdown valves CLOSED • Steam supply to TD AFW pump ISOLATED (S/G 1 or 4). **IF** any Faulted S/G **NOT** isolated, **THEN** **GO TO** E-2, Faulted Steam Generator Isolation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**CHECK** S/G tube integrity: a. **CHECK** the following indications of a S/G tube rupture, including available trends prior to isolation: • Any S/G level rising in an uncontrolled manner. **OR** • Main steamline high radiation. **OR** • Condenser exhaust high radiation. **OR** • S/G blowdown recorder RR-90-120 , pen #1 and pen #2 high radiation. **OR** • Post-Accident Area Radiation Monitor recorder RR-90-268B, points 3 (blue), 4 (violet), 5 (black), or 6 (brown) high radiation. [M-31 (back of M-30)]	**PERFORM** the following: 1) **VERIFY** procedure in effect is one of the following E-1 or ECA-1 series procedures: • E-1, Loss of Reactor or Secondary Coolant. • ES-1.1, SI Termination. • ES-1.2, Post LOCA Cooldown and Depressurization. • ES-1.3, Transfer to RHR Containment Sump. • ES-1.4, Transfer to Hot Leg Recirculation. • ECA-1.1, Loss of RHR Sump Recirculation. • ECA-1.2, LOCA Outside Containment. 2) **IF** E-1 or ECA-1 series procedure in effect, **THEN** **RETURN TO** procedure and step in effect. 3) **IF** E-1 or ECA-1 series procedure **NOT** in effect, **THEN** **GO TO** E-1, Loss of Reactor or Secondary Coolant.

77. FH-B.12.B 001

Which ONE ~~(1)~~ of the following describes the function of the SFP bridge crane hoist "lower" limit gear operated switch?

✓A. Stops downward travel before the crane hook enters the water.

B. Stops downward travel before the fuel assembly reaches the bottom of the fuel cell.

C. Automatically changes hoist speed from fast to slow while the fuel assembly is in the fuel cell boundary.

D. Prevents bridge travel when the fuel assembly is within the fuel cell boundary.

A. Correct. This prevents contamination of the hook and possible contamination of SFP water.
B. Fuel handling operator uses a load cell to monitor and stop downward travel when the FA reaches the bottom of the fuel cell.
C. There is no fast/slow speed on the hoist. It only has a normal speed with a jog button
D. Bridge travel is interlocked with the full up limit switch.

K/A{CFR}:	036 AK3.02	[2.9-3.6]	{41.5, 41.10}
	034 K4.02	[2.5/3.3]	{41.7}
	034 A3.01	[2.5/3.1]	{14.7}

References: FHI-3 Sect. D. IV.A

LP/Objectives: System Description 079a Obj. 12.b

History: Refuel.bnk Q# OPL274A069.5 004. Revised stem and D distractor.

Level: Memory

Comments: LP-5/2000.; FH 001; FHW 12/02 034 K4.02

A4.02 Better

Unit______________ Date___________

IV. INTERLOCKS AND BYPASS SWITCHES

A. SFP Bridge Hoist Interlocks/Limit Switches:

1. The hoist is interlocked with the bridge crane such that the hoist will not move up or down if the bridge is moving in either direction.

2. The bridge is interlocked with the hoist such that the bridge can not move in normal speed if the hoist is not full up (jog would be available).

3. The lower limit switch is a gear operated switch that will activate to stop hoist travel before the hook enters the water.

4. The upper limit is also a gear operated switch that will stop upward travel before the hook reaches the hoist drum block assembly.

5. The "hoist overtravel rod" is a backup mechanical-electrical stop that stops upward travel when the hook hits the rod just below the hoist drum block assembly.

B. Transfer Cart/Upender Interlocks:

1. Valve Open - Wafer valve has to be fully open before any traverse can begin - cannot be bypassed with a switch.

2. Frame Down - Both frames have to be down before traverse can begin - can be bypassed with a switch.

3. Conveyor at Pit - Conveyor has to be at the SFP side before upending can begin on the SFP side - can be bypassed with a switch.

4. Conveyor at Rx - Conveyor has to be at the Rx side before upending can begin on the Rx side - can be bypassed with a switch.

5. Manipulator Crane - The manipulator crane has to be over the core or the gripper tube is fully up - cannot be bypassed with a switch.

76. ES-1.2-B.2 006

During the performance of ES-1.2, "Post-LOCA Cooldown and Depressurization," it is desirable to have only one RCP running.

Which ONE ~~(1)~~ of the following describes the reason for having only one RCP in service?

A. One RCP provides the DELTA-P required to provide letdown. Additional RCPs would add unnecessary heat load.

✓B. One RCP is desired for spray and RCS heat transport to the SGs. Additional RCPs would add unnecessary heat load.

C. One RCP is needed for RCS heat transport to the SGs. Additional RCPs could overload the electrical power supply.

D. One RCP is desired for spray and RCS mixing. Additional RCPs would strain the plant electrical power supply in the post-LOCA condition.

A. Incorrect - RCS pressure provides force for letdown.
B. Correct - per EPM-3 "Basis Document"
C. Incorrect - RCPs are fed from independent 6.9 Kv Unit Boards and will not overload them.
D. Incorrect - RCPs are fed from independent 6.9 Kv Unit Boards and will not overload them.

K/A {CFR}:	W/E03 EK3.1	[3.3/3.7]	{41.5, 46.13}
	E03 EA1.2	[3.7/3.9]	{41.7}

References: EPM-3-ES-1.2. Step 14

LP/Objectives: OPL271C387 B.2

History: Procedure Bank

Level: Comprehension

Comments: FHW 12/02 E03 EA1.2

EK 3.1 much better

EOP Step Number: 14

DETERMINE if one RCP should be started:

ERG Step Number: 12

Check If An RCP Should Be Started:

Purpose:

To establish forced circulation flow from one RCP.

ERG Basis:

Forced coolant flow is the preferred mode of operation to allow for normal RCS cooldown and provide pressurizer spray. If RCPs had not been tripped, all but one are now stopped to minimize heat input to the RCS. The RCP started or left running should be one that can provide normal pressurizer spray (see preceding note). If no RCP is running, RCS subcooling, pressurizer level, and certain plant specific conditions are required before starting an RCP.

Depressurization of the RCS may generate a steam bubble in the upper head region of the reactor vessel if no RCP is running. This bubble could rapidly condense during pump startup, drawing liquid from the pressurizer and reducing reactor coolant subcooling. If pressurizer inventory is not sufficient, level may decrease off span. In addition, local flashing of reactor coolant could occur if RCS subcooling is not adequate. These conditions would require SI reinitiation and may confuse the operator if such behavior was unexpected.

Knowledge:

Plant-specific procedures for starting an RCP may require a steam bubble to be present in the pressurizer. RCP restart should be permitted if an RCS leak path is certain since the leak ensures that there will not be a significant pressure surge when the RCP is started.

EOP Basis:

Same.

75. ES-1.2-B.2 004

Given the following plant conditions:

- A LOCA has just occurred on Unit 1 and appropriate responses have been taken.
- Only #2 RCP is running.
- The crew is commencing post-LOCA cooldown and depressurization.
- ~~ES-1.2, Post LOCA Cooldown and Depressurization, contains cautions relative to RCS voiding.~~

Which ONE ~~(1)~~ of the following is an indication that voiding exists in the RCS?

A. Rapidly increasing RVLIS Upper Plenum level if auxiliary spray is initiated.

B. Rapidly decreasing safety injection flow rate when RCS pressure is decreased.

✓C. Rapidly increasing pressurizer level when normal spray valves are opened.

D. Rapidly decreasing RCS pressure if high head injection is realigned through the CCPIT.

A. Incorrect, auxiliary spray would cause pressurizer level to increase.
B, Incorrect, decreasing RCS pressure would allow increased SI flow.
C. Correct per ES-1.2.
D. Incorrect, SI flow throught the CCPIT would increase RCS pressure.

K/A {CFR}:	000008K301 [3.7/4.4] E03 EK1.3 [3.5/3.8] {41.8 41.10}
References:	ES-1.2, caution at step 13.
LP/Objectives:	OPL271C387, B.2
History:	HLC 9809 Audit Exam, HLC 12/02
Level:	Memory
Comments:	FHW 12/02 E03 EK1.3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**CHECK** if ECCS in service: • Any SI pump RUNNING. **OR** • CCPIT flow INDICATED. **OR** • Any RHR pump RUNNING in SI mode.	**GO TO** Step 20.
12.	**TURN OFF** pressurizer heaters.	

CAUTION **Upper head voiding due to RCS depressurization may result in rapidly rising pressurizer level.**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**DEPRESSURIZE** RCS to refill pressurizer:	
	a. **USE** normal pressurizer spray.	a. **USE** one pressurizer PORV. **IF** pressurizer PORV NOT available, **THEN** **USE** auxiliary spray **USING** EA-62-4, Establishing Auxiliary Spray.
	b. **INITIATE** RCS depressurization.	
	c. **WHEN** pressurizer level greater than 20% [35% ADV], **THEN** **STOP** RCS depressurization.	

72. ES-0.1-B.1 002

A reactor trip following a spurious turbine runback occurred on Unit 1. The following plant conditions are observed:

- RCS Tavg = 547° F
- RCS pressure = 2198 psig and slowly increasing
- CCP 1 A-A tripped on overcurrent at the time of the trip
- Control Rods are fully inserted except Rods H-8 and C-4 are at 24 and 16 steps respectfully.

Which ONE ~~(1)~~ of the following describes the appropriate action for the OATC to take?

A. Start CCP 1B-B, open FCV-62-132 and FCV-62-133 (VCT Outlet), close FCV-62-135 and FCV-62-136 (RWST Supply to CCP Suction), verify at least 80 gpm charging flow through the normal charging flowpath.

✓B. Start CCP 1B-B, shift the BAT pumps to fast, open FCV-62-138 (Emergency Borate), verify at least 35 gpm boric acid flow.

C. Shift the BAT pumps to fast, open FCV-62-138 (Emergency Borate), verify at least 35 gpm boric acid flow.

D. Shift the BAT pumps to fast, open FCV-62-140 (Normal Boration to VCT Outlet) and FCV-62-128, verify at least 30 gpm boric acid flow.

A. incorrect this is normal boration path.
B. correct per EA-68-4 requires emergency boration.
C. incorrect a CCP is required for emergency boration.
D. incorrect this is not an emergency boration path.

K/A {CFR}:	024 AK3.02		
	024 AA1.26	[3.3/3.3]	{41.7}
References:	EA-68-4		
	ES-0.1		
LP/Objectives:	OPL271C381, b.1		
History:	Procedure Bank		
Level:	Comprehension		
Comments:	FHW 12/02 024 ~~AA1.26~~		

AA 1.17

4.1 **Section Applicability (Continued)**

3. **IF** entering this instruction from ES-0.1 due to two or more control rods indicating greater than 12 steps,
   THEN
   PERFORM the following:

   a. **IF** using BAT as boration source,
      THEN
      GO TO Section 4.2, Emergency Boration from BAT. ☐

   b. **IF** using RWST as boration source,
      THEN
      GO TO Section 4.3, Emergency Boration from RWST. ☐

4. **IF** Emergency Boration to be terminated,
   THEN
   GO TO Section 4.4. ☐

5. **RETURN TO** procedure and step in effect. ☐

END OF SECTION

4.2 Emergency Boration from the BAT

1. **PLACE** boric acid transfer pumps in fast speed. ☐

2. **ADJUST** emergency borate valve **[FCV-62-138]** to obtain boric acid flow between 35 gpm and 150 gpm on **[FI-62-137A]**. ☐

3. **MONITOR** emergency boration flow:

   a. **CHECK** emergency boration flow established on **[FI-62-137A]**. ☐

   b. **IF** boric acid flow less than 35 gpm,
   THEN
   CLOSE recirculation valve for the BAT aligned to the blender: ☐

   - 1-FCV-62-237 for BAT A.
   - 0-FCV-62-241 for BAT C.
   - 2-FCV-62-237 for BAT B.

4. **IF** emergency boration flow NOT established,
THEN
ALIGN normal boration path:

   a. **VERIFY** VCT outlet valves LCV-62-132 and LCV-62-133 OPEN. ☐

   b. **ALIGN** normal boration to VCT outlet:

   - **OPEN** FCV-62-140. ☐
   - **OPEN** FCV-62-144. ☐

   c. **CHECK** boration flow greater than 35 gpm on FI-62-139. ☐

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	g. Total feed flow to S/Gs greater than 440 gpm.	g. **ENSURE** AFW operation **USING** EA-3-8, Manual Control of AFW Flow. **IF** AFW can **NOT** be established, **THEN** **ESTABLISH** MFW pump operation and MFW flow control on MFW regulating bypass valves.
NOTE	A control rod is considered fully inserted if it is less than or equal to 12 steps from bottom.	
5.	**VERIFY** all control rods fully inserted. • Rod bottom lights LIT. • Rod position indicators less than or equal to 12 steps.	**IF** two or more RPIs indicate greater than 12 steps, **THEN** **EMERGENCY BORATE USING** EA-68-4, Emergency Boration, to compensate for loss of shutdown margin due to stuck control rod(s).
6.	**ANNOUNCE** reactor trip **USING** PA system.	

79. FR-C.1-B.2 003

Given the following plant conditions:

- The Unit has tripped from 100% power with a LOCA in progress.
- Pressurizer pressure indicates 900 psig.
- Reactor Coolant Pumps have been tripped.
- Core exit thermocouples are at 720°F.
- RVLIS lower plenum indicates 36%.
- Tavg for All loops range between 512°F - 525°F

Which ONE ~~(1)~~ of the following describes the conditions existing in the core as applicable to the Emergency Operating Procedures?

✓A. Superheated conditions, which present an imminent challenge to the fuel matrix and fuel cladding.

B. Superheated conditions, which do **NOT** present a challenge to the fuel matrix and fuel cladding as long as hot leg temperatures are at saturation conditions.

C. Saturated conditions, which present ~~a~~ potential challenge to the fuel matrix and fuel cladding.

D. ~~Saturated~~ subcooled conditions, which do **NOT** present a challenge to the fuel matrix and fuel cladding.

A. Correct, per steam table. RCPs off and RVLIS < 40 % equates to core water level < 3.5 feet from bottom of the fuel, therefore, imminent challenge is correct.
B. Incorrect, water level is < 3.5 feet from bottom of the fuel.
C. Incorrect, conditions are superheat.
D. Incorrect, conditions are superheat and water level is < 3.5 feet from bottom of the fuel.

K/A {CFR}: 074 EK1.01 [4.3/4.7] {41.8 41.10}

References:
FR-C.1
EPM-3-FR-C.1

LP/Objectives: OPL271C398, B.2

History: HLC 9809 Audit Exam
HLC 12/02

Level: Comprehension

Comments: **Provide steam tables.** FHW 12/02 074 EK1.01 ✓

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**CHECK** RVLIS lower range indication:	
	a. **CHECK** indication greater than 40%.	a. **IF** rising, **THEN** **GO TO** Step 2. **IF** stable or dropping, **THEN** **GO TO** Step 10.
	b. **RETURN TO** procedure and step in effect.	

EOP Step Number: 9

CHECK RVLIS lower range indication:

ERG Step Number: 6

Check RVLIS Full Range Indication:

Purpose:

To check if an RCS inventory condition symptomatic of an inadequate core cooling condition still exists.

ERG Basis:

The trend in RVLIS full range indication is used to check the effectiveness of safety injection in restoring RCS inventory. If increasing, then no further action may be necessary. The operator is instructed to return to Step 1 and repeat the initial guideline steps until the RVLIS full range indication is greater than (3).

If the RVLIS full range indication is greater than (3), then safety injection has been successful in restoring RCS inventory and core cooling. This step will transfer the operator to the guideline and step in effect.

If the RVLIS full range indication is not increasing, then the operator is instructed to check core exit TCs in the next step to determine if an inadequate core cooling condition still exists.

EOP Basis:

Same.

Deviation:

RVLIS full range changed to RVLIS lower range.

Justification:

AT SQN, RVLIS lower range monitors vessel level in the active fuel region.

Setpoint:

Identifier: <J01>
Description: RVLIS value which is 3.5 feet above the bottom of active fuel in core with zero void fraction - normal accuracy.

K/A ans

80. FR-C.1-B.2 011

Given the following plant conditions:

- The operating crew entered procedure FR-C.1, "Inadequate Core Cooling."
- All attempts to establish high pressure Safety Injection flow were unsuccessful.
- RVLIS lower range level is 28% and dropping slowly.
- Core Exit Thermocouples are reading 820 degrees F and slowly increasing.
- Reactor Coolant pumps have been secured.

Which ONE ~~(1)~~ of the following methods would be the **NEXT** step in mitigating the core cooling challenge?

A. Enter the Severe Accident Management Guidelines (SAMGs).

B. Prior to RCP restart open available pressurizer PORVs to allow RCS depressurization to the SI accumulator and SI injection pressures.

✓C. Depressurize all intact steam generators using Steam dumps or ARVs to allow RCS depressurization to the SI accumulator and SI injection pressures.

D. When RCP support condtions are available restart one RCP in an idle loop to provide forced two-phase flow through the core.

a. Incorrect - This event does not require entry into SAMGs
b. Incorrect - If RCPs cannot be started, or if core cooling is not restored after the RCPs are started, then RCS vent paths are opened. RCS vent paths are not considered prior to restart of RCP.
c. Correct -This is first evolution attempted if normal ECCS is not established
d. Incorrect - If the secondary depressurization is not successful in reducing core temperatures, then the RCPs are started. Support conditions are not required to start a RCP.

Neither K/A matches

EK 2.06

80. FR-C.1-B.2 011

K/A {CFR}: EPE 074 EK2.05 [3.9/4.1] {41.7, 45.7}
074 EK2.02 [3.9/4.0] {41.7}

References: FR-C.1 "Inadequate Core Cooling" step 14

LP/Objectives: OPL271C398 B.2

History: Bank; Modified distractor B to include "prior to RCP restart" to avoid confusion as to assumed procedure step progress. Modified distractor D to include "when RCP support conditions are available" to avoid confusion as to assumed procedure step progress. FHW 12/02

Level: Analysis

Comments: FHW 12/02 074 EK2.02
EK2.06

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- Blocking low steamline pressure SI as soon as pressurizer pressure is less than 1920 psig will prevent an inadvertent MSIV closure and keep the condenser available for steam dump.
- After the low steamline pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate is exceeded.
- S/G depressurization at the maximum rate may cause S/G narrow range levels to drop to less than 10% [25% ADV]. This is acceptable and expected for this inadequate core cooling condition.

14. **DEPRESSURIZE** Intact S/Gs to reduce RCS pressure to less than 125 psig:

    a. **WHEN** RCS pressure less than 1920 psig,
    THEN
    PERFORM the following:

    1) **BLOCK** low steamline pressure SI.

    2) **CHECK** STEAMLINE PRESS ISOL/SI BLOCK RATE ISOL ENABLE permissive LIT. [M-4A, A4]

    b. **DUMP** steam to condenser at maximum rate.

    Response Not Obtained:

    b. **DUMP** steam at maximum rate **USING** Intact S/G atmospheric relief(s).

    IF local control of atmospheric relief(s) is necessary,
    THEN
    DISPATCH personnel to dump steam **USING** EA-1-2, Local Control of S/G PORVs.

(Step continued on next page.)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	c. **CHECK** RCS pressure less than 125 psig.	c. **IF** dropping, **THEN** **GO TO** Caution prior to Step 12. **IF** stable or rising, **THEN** **GO TO** Note prior to Step 21.
	d. **STOP** S/G depressurization.	
15.	**MONITOR** if CLAs should be isolated:	
	a. **CHECK** RCS pressure less than 125 psig.	a. **GO TO** Note prior to Step 21.
	b. **RESET** SI and **CHECK** the following: • AUTO S.I. BLOCKED permissive LIT. [M-4A, C4] • S.I. ACTUATED permissive DARK. [M-4A, D4]	
	c. **CLOSE** CLA isolation valves.	c. **VENT** any unisolated CLA **USING** EA-63-1, Venting Unisolated Cold Leg Accumulator.

Rewrite

81. FR-H.1-B.3 004

The emergency procedures have been implemented and the following plant conditions exist:

1. Performance of FR-H.1, "Loss of Secondary Heat Sink", is in progress.
2. Neither Feedwater or Condensate flow can be established.

Which ONE ~~(1)~~ of the following ~~feed and bleed~~ methods will be used?

A. Feed makeup from the CVCS to the Reactor Coolant System and use the Pressurizer PORV's to bleed off the decay-heat energy.

✓B. Feed ECCS to the Reactor Coolant System and use the Pressurizer PORV's to bleed off the decay-heat energy.

C. Feed makeup from the CVCS to the RCS and use normal Letdown to bleed the RCS.

D. Feed ECCS to the RCS and use the Steam-Generator ARV's to bleed steam to the atmosphere.

A. Incorrect, procedure requires actuation of SI to start ECCS pumps to feed.
B. Correct per FR-H.1.
C. Incorrect, procedure requires actuation of SI and bleeds using PORVs.
D. Incorrect, procedure requires using PORVs for a bleed path.

K/A[CFR]: E05EK2.2 (3.9-4.2)
E05 EA1.3 [3.8/4.2] {41.7}

Reference: FR-H.1

LP/Objective: OPL271C401, b.3

History: Procedure bank old Bank Number PL-0900

Level: ~~Comprehension~~ memory

Comments: FHW 12/02 E05 EA1.3

Bleed & Feed vice Feed & Bleed

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17.	**VERIFY** RCS feed path: a. **CHECK** ECCS pump status: • At least one CCP RUNNING **OR** • At least one SI pump RUNNING.	**START** pumps and **ALIGN** valves as necessary. **IF** a feed path **CANNOT** be established, **THEN** **CONTINUE** attempts to establish feed flow. **GO TO** Step 10.
	b. **CHECK** ECCS valves ALIGNED as appropriate: • **REFER TO** EA-63-5, ECCS Injection Mode Alignment. **OR** • **REFER TO** ES-1.3, Transfer to RHR Containment Sump. **OR** • **REFER TO** ES-1.4, Transfer to Hot Leg Recirculation.	
18.	**ESTABLISH** RCS bleed path:	
	a. **CHECK** power to pressurizer PORV block valves AVAILABLE.	a. **DISPATCH** personnel to restore power to block valves **USING** EA-201-1, 480 V Board Room Breaker Alignments.
	b. **CHECK** pressurizer PORV block valves OPEN.	b. **OPEN** block valves.
	c. **OPEN** pressurizer PORVs.	

82. FR-P.1 001

FR-P.1, Response to Pressurized Thermal Shock, has the operator check that RHR flow is greater than 1500 gpm if RCS pressure is less than 180 psig.

This step is based on:

A. ensuring adequate mixing in the cold leg downcomer region during natural circulation conditions.

B. preventing core exit temperatures from exceeding the required temperature to place RHR in service.

C. ensuring adequate Low Head Safety Injection cooling prior to Accumulator isolation.

✓D. preventing implementation of Pressurized Thermal Shock (PTS) actions if a Large Break LOCA has occurred.

A. Incorrect, RHR does not mix the CL region; RCS is uncoupled.
B. Incorrect, CET has no input to RHR
C. Incorrect, Isol CLA in P.1
D. Correct per EPM-3-FR-P.1

K/A: E08 EK3.2 [3.6/4.0] {41.5 41.10}

Reference: EPM-3-FR-P.1

Objective: OPL271C406 b.3

History: Procedure Bank

Level: Comprehension

Comments: FHW 12/02 E08 EK3.2✓

EOP Step Number: 3

CHECK RCS pressure greater than <B08> psig.

ERG Step Number: N/A

Purpose:

To determine if the entry into FR-P.1 was due to a large-break LOCA.

ERG Basis:

For transients where RCS pressure is less than the RHR pump shutoff head and flow from the RHR pumps has been verified, the operator should return to the guideline and step in effect since these symptoms are indicative of a large-break LOCA. In this instance, the actions in FR-P.1 should not be performed since pressurized thermal shock is not a serious concern for a large-break LOCA.

A large-break LOCA will depressurize the RCS to containment pressure in a matter of minutes. RCPs will be tripped due to low pressure and natural circulation will stop as fluid levels in the S/G tubes drop below the U-bends. A safety injection signal will be actuated by low pressurizer pressure. Large volumes of water will be injected into the loop cold legs, leading directly to the severe cooling of the vessel downcomer. Most of the water will come from the RWST, which is at the ambient temperature of its location; this may be either indoors or outdoors, so water temperature could range from 70°F down to near freezing. Another source of water injected into the cold legs comes from pressurized accumulators located inside containment. Depending on containment design and power history, ambient temperature could range from 70°F to 120°F. This relatively cold safety injection water produces significant thermal shock in the downcomer region, but for the large-break LOCA, pressure is not a concern. Most of the water initially in the RCS, and most of the colder water added by injection, will flow out the break. This break flow mixes together in the containment sump to result in a somewhat higher temperature for the sump water than that purely from injection. During the recirculation phase of recovery, this warmer fluid is pumped back into the RCS, somewhat offsetting the initial thermal shock.

As discussed in the Pressurized Thermal Shock (PTS) Training Program documentation, temperature stress alone is normally not sufficient to threaten the integrity of the pressure vessel. The PTS scenario also assumes that the RCS is pressurized and a critical flaw which is the correct size, shape, and orientation exists at the inner surface of the vessel wall. In addition, the warm prestressing effect, which results in an apparent increase in fracture toughness of the vessel material, is applicable to this situation since repressurization of the RCS is virtually impossible following a large-break LOCA. Any material flaw would be prevented from reinitiating and growing beyond 75% of the vessel wall thickness. Therefore, for a large-break LOCA, PTS is not a serious concern and Guideline FR-P.1 does not need to be implemented. (Reference DW-92-032).

EOP Basis:

Same. The value for <B07> and <B08> (corresponding to the ERG normal and adverse containment setpoints) has been calculated to be the same. Therefore, only one value is displayed for these setpoints. Refer to EPM-6, EOI Setpoints Document, for additional information.

Deviation:

None.

Justification:

N/A.

Setpoint:

Identifier: <B07>
Description: Shutoff pressure for RHR pumps - normal accuracy.

Identifier: <B08>
Description: Shutoff pressure for RHR pumps - adverse containment accuracy.

Identifier: <S03>
Description: Minimum RHR pump flow to indicate injection into the RCS.

K/A MM

83. FR-Z.1-B.2 001
FR-Z.1, "High Containment Pressure", directs that if ECA-1.1, "Loss of RHR Sump Recirculation", is in effect then operate containment spray using applicable steps in ECA-1.1.

Which ONE ~~(1)~~ of the following describes the basis for giving priority to ECA-1.1?

A. ECA-1.1 operates the containment spray pumps to maintain level in the containment sump for the RHR pumps.

B. ECA-1.1 operates the containment spray pumps to ensure ~~sufficient power is available from the diesel generators for the RHR pumps.~~ → ID containment pressure is the first priority in the mitigation strategy.

C. ECA-1.1 operates the containment spray pumps to prevent automatic swapover of the spray pumps to the containment sump.

✓D. ECA-1.1 operates the containment spray pumps in order to conserve the level in the RWST.

A. Incorrect, not addressed in EPM-3-ECA-1.1.
B. Incorrect, not addressed in EPM-3-ECA-1.1.
c. Incorrect, not addressed in EPM-3-ECA-1.1.
D. Correct, the RHR sump is not available, therefore the crew must conserve water for injection from the RWST.

K/A: 000069SG11 [3.8 / 4.2]
025 AK2.05 [2.6/2.6] {41.7}

Reference: EPM-3-ECA-1.1 step 12

LP/Objective: OPL271C417, B.2

History: Procedure Bank

Level: ~~Comprehension~~ memory

Comments: FHW 12/02 025 AK2.05

AK 3.01 Better K/A

EOP Step Number: 12

MONITOR RWST level greater than <U02>%.

ERG Step Number: 8

Check RWST Level - GREATER THAN (1)

Purpose:

To determine course of action based upon amount of water available in RWST.

ERG Basis:

If the RWST is not empty, the operator proceeds with Steps 9 through 26, which are concerned with minimizing the RWST outflow and, therefore, extending the time that fluid for core cooling is provided by the RWST. This is accomplished by stopping the containment spray pumps and decreasing the SI pump flowrate. However, if the RWST is empty, the operator is instructed to skip to Step 27. In Step 27, the operator stops all pumps taking suction from the empty RWST.

EOP Basis:

Same.

Deviation:

The EOP step is changed to a continuous action step.

Justification:

In accordance with DW-92-056.

Setpoint:

Identifier: <U02>
Description: RWST Lo-Lo alarm (empty).

84. FR-Z.2-B.2 001
Which ONE ~~(1)~~ of the following is addressed in FR-Z.2, "Containment Flooding", as the potential source of excessively high containment sump levels?

A. Condensed steam from a steam break

B. RWST

C. Accumulators

✓D. CCS

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

Student must comprehend the RWST and Accumulators are bounded [limited] sources and the CCS has potential to continue flowing. Also the engineering evaluation for the sump level transmitter setpoint [not high level] includes the volume of the RWST and the accumulators.
K/A{CFR}: E15 EK1.1 [2.8/3.0] [41.8 41.10]

Reference: FR-Z.2

LP/Objective: OPL271C409, b.2

History: Old Bank Number PL-0014

Level: Memory

Comments: FHW 12/02 E15 EK1.1 ✓

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. **ATTEMPT** to identify and isolate unexpected source of water to containment sump:

   - ERCW
   - CCS
   - High pressure fire protection
   - Primary water
   - Demineralized water
   - SFP cooling water.

2. **CHECK** containment sump activity level:

   a. **CHECK** RHR suction ALIGNED to RWST.

   Response Not Obtained: a. **NOTIFY** chem lab to sample RHR System for activity.
   GO TO Step 3.

   b. **NOTIFY** chem lab to sample containment sump for activity.

88. INCORE-B.1.B 002

The plant Exosensor System calculates RCS subcooling by using which ONE ~~(1)~~ of the following instrument inputs?

A. Pressurizer pressure and RCS WR T_{hot}.

B. Pressurizer pressure and core exit thermocouples.

C. RCS WR pressure and RCS WR T_{cold}

✓D. RCS WR pressure and core exit thermocouples.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A {CFR}: 016 K1.01 [3.4/3.4] [41.2-9]

References: GOI-6 section P

LP/Objectives: OPL271RCSTEMP, B.9

History: System bank

Level: Memory

Comments: FHW 12/02 016 K1.01 Exosensor also uses non nuclear inputs for RCS subcooling calculation, i.e. RCS WR pressure.

INCORE THERMOCOUPLE TEMPERATURE AND SUBCOOLING MARGIN MONITORING SYSTEM (EXOSENSOR TEMPERATURE MONITOR)

1.0 INTRODUCTION

The Incore Thermocouple Temperature and Subcooling Margin Monitoring System provides fuel assembly outlet temperatures at selected core locations and calculates saturation margin. The system consist of two (2) 1E channels which are safety-related, designed, and qualified for post-accident monitoring (PAM). The principle function is to provide information to the operator during an accident situation for assessing plant post-accident status and verify core-cooling and subcooling conditions, as well as detection of potential fuel-clad breech following an accident. The system is used for acquiring data only, and performs no operational plant control. The operator has the capability to select thermocouples that are to be monitored by the Incore Thermocouple Temperature and Subcooling Margin Monitoring System.

The Incore Thermocouple Temperature and Subcooling Margin Monitoring System uses the hottest incore thermocouple, the hottest wide-range hot leg RTD, average of all thermocouples, and RCS wide-range pressure for calculations.

2.0 SYSTEM DESCRIPTION

A. Microprocessor Inputs

1. Channel 1 Microprocessor receives analog voltage signals from 34 thermocouples, 3 cold-junction reference RTD's, 2 wide-range hot-leg RTD's, and 1 wide-range pressure transmitter. Power supply is from the Vital Inst. Pwr. Bd. 1-I (Unit 1) and 2-I (Unit 2).

2. Channel 2 Microprocessor receives analog voltage signals from 31 thermocouples, 3 cold-junction reference RTD's, 2 wide-range hot-leg RTD's, and 1 wide-range pressure transmitter. Power supply is from the Vital Inst. Pwr. Bd. 1-II (Unit 1) and 2-II (Unit 2).

B. Key Pad layout:

TC TEMPS 1	TC LIMIT 2	REF RTD 3	DELETED RTD/TC 4	MARGIN 5	SELECT TC	ESC	ALARM
RCS 6	QUAD 1 7	QUAD 2 8	9	QUAD 3 0	QUAD 4	NEXT	HELP ENTER

89. INCORE-B.1.D 003

The post-accident monitoring instrumentation for core-exit temperature and saturation margin is located on M4 Control Panel. One of the Main Menu displays on the Channel-I plasma display monitor has an asterisk (*) notation.

Which ONE ~~(1)~~ of the following describes the current alarm ~~at the bottom of the monitor,~~ which is associated with the asterisk?

✓A. "Any (core-exit) Thermocouple ~~Above~~ outside Physical Limit." A thermocouple is above or below its limit setpoint.

"Denotes Thermocouple outside of electrical limits."

B. ~~"Inadequate Saturation Temperature Margin." Margin is <40 degrees~~.

C. "Monitor Mode is in Effect." The channel has been taken off line for calibration from the Auxiliary Instrument Room panel.

D. "Diagnostic Channel Error." All capability for calculating Saturation Margin has been disabled.

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.
K/A: 017 K6.01 [2.7-3.0] [41.7]

Reference: Incore Temperature Monitoring System Description section 2 "Plasma Display Unit." GOI-6 section P

LP/Objective: Incore Temperature Monitoring System Description Objective 3.

History: Old Bank Number PL-1126

Level: Memory

Comments: FHW 12/02 017 K6.01

Describe what you mean by "alarm at the bottom of the monitor?" Does it display words. If not then ~~take~~ remove quotes.

C. Key description:

KEY LABEL	DISPLAY TITLE	DISPLAY DESCRIPTION
TC Temps	Thermocouple (TC) Temperatures	Displays average, hottest, and current user selection thermocouple and trend parameters.
TC Limits	TC Limits	Gives upper and lower limits for each TC number.
Ref RTD	RTD Temperature and Physical Limits	Summarizes RTD reference-junction temperatures and physical limits (low and high).
Deleted RTD/TC	Deleted TC and RTD Sensors by Tag No.	Shows by Tag number TC's and RTD's that have been deleted.
Margin	Margin to Saturation	Summarizes and displays all six calculations for margin to saturation.
Select TC	Select Crnt/Quad TC, Trend Recorders	Lists a menu of sub displays for current TC, TC Quadrants, and Recorders.
ESC	Escape	Allows user to quit what they are doing and go back to the menu/display.
Alarm	Alarm Status	Displays current alarm status plus all accumulated alarms.
RCS	RCS Temperatures and Pressures	Gives RCS Temperatures and Pressures.
Quad #1	Quad #1	Lists Temperature of TC's in Quadrant #1
Quad #2	Quad #2	Lists Temperature of TC's in Quadrant #2
9	No Display	Not Connected
Quad #3	Quad #3	Lists Temperature of TC's in Quadrant #3
Quad #4	Quad #4	Lists Temperature of TC's in Quadrant #4
Next	Next	Allows user to inspect/modify or step through the next item in a long list.
Help Enter	Help	Tells operator what action to take. Can only exit by depressing the ESC key.

D. Plasma Display Unit/Indicator (PDU)

The PDU provides an indication of monitored plant parameters status.

1. Display Notations:

*	Denotes TC outside of physical limits
?	Denotes TC outside of electrical limits
XXX	Denotes a deleted TC

91. INPO2 976

Given the following conditions:

- A Reactor trip and Safety Injection have occurred due to a SGTR.
- E-3, "Steam Generator Tube Rupture," is in effect with a cooldown started at maximum rate. to Target core exit temperature.
- The highest S/G pressure is 900 psig.
- RCS pressure is 1000 psig and dropping.
- High Head Safety Injection flow is approximately 300 gpm.

Which of the following describes what should be done with the Reactor Coolant Pumps?

A. RCP's should must be tripped because RCP trip criteria is currently met.

✓B. RCP's should must **NOT** be tripped because RCP trip criteria does **NOT** apply once an operator initiated RCS cooldown is commenced.

C. RCP's should must be tripped because RCP trip criteria applies after an operator initiated RCS depressurization is commenced.

D. RCP's should must **NOT** be tripped because RCP trip criteria does **NOT** apply until the operator initiated RCS cooldown is completed.

A. Incorrect per procedure.
B. Correct per procedure.
C. Incorrect per procedure.
D. Incorrect per procedure.

K/A [CFR]:	038 EK3.08 [4.1/4.2] {41.5 41.10}
References:	E-3
LP/Objective:	OPL271C391 B.3
History:	INPO2 976
Level:	Comprehension
Comments:	FHW 12/02 038 EK3.08

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- Blocking low steamline pressure SI as soon as pressurizer pressure is less than 1920 psig will prevent an inadvertent MSIV closure and keep the condenser available for steam dump.
- After the low steamline pressure SI signal is blocked, main steamline isolation will occur if the high steam pressure rate setpoint is exceeded.
- The 1250 psig RCP trip criterion is NOT applicable after RCS cooldown is initiated in the following step.

9. **INITIATE** RCS cooldown:

   a. **DETERMINE** target core exit T/C temperature based on Ruptured S/G pressure:

Lowest Ruptured S/G pressure (psig)	Target Core Exit T/C Temp (°F)
1100	515
1000	500
900	490
800	475
700	460
600	445
500	425
400	400
380	390

(Step continued on next page.)

92. INPO3 955

Given the following plant conditions:

- A small break LOCA has occurred.
- SI pumps fail to start.
- RCS Hot Legs and the Reactor Vessel Head have voided.
- RCS Pressure is 775 psig.
- RCP's are tripped in accordance with the EOP network.
- Assume that all other ECCS equipment operates as required.

Which ONE (X) of the following describes the current method of cooling the core?

A. Break flow is the only core cooling method available.

B. Natural Circulation is the principle means of core cooling.

C. No core cooling mechanism exists at the present time.

✓D. Break flow and reflux flow are providing core cooling.

A. Incorrect, the initial conditions due not exclude a S/G heat sink.
B. Incorrect, voiding in the hot legs and vessel head does not allow natural circulation.
C. Incorrect, a heat sink and break flow cooling still exists.
D. Correct, initial conditions would allow break and reflux flow.

K/A [CFR]: 009 EK1.01 [4.2/4.7] {41.8 41.10}

References: EPM-3-ES-0.3 "Check RVLIS upper plenum range."

LP/Objectives: OPL271C383 b.4

History: INPO3 955

Level: Comprehension

Comments: FHW 12/02 009 EK1.01

EOP Step Number: 7

CHECK RVLIS upper plenum range greater than <I02>%.

ERG Step Number: 5

Check RVLIS Upper Range Indication - GREATER THAN (5).

Purpose:

To ensure that the void does not enter the hot legs and disrupt natural circulation.

ERG Basis:

If steam enters the hot legs, it would most likely be condensed by the subcooled hot leg fluid. However, there may be the potential for some steam to enter the hot legs and reach the top of the S/G U-tubes, thereby disrupting the natural circulation flow circuit. By monitoring RVLIS and limiting the void growth to the top of the hot legs (repressurizing the RCS if necessary), the potential for introducing voids into the S/G U-tubes is minimized. An uncertainty is applied to the reading to ensure that the void will actually be above the hot legs, even when assuming worst case channel inaccuracies.

EOP Basis:

Same.

Deviation:

None.

Justification:

N/A

Setpoint:

Identifier: <I02>
Description: RVLIS value corresponding to the top of the hot legs - normal accuracy.

93. INPO8 155

When synchronizing the Main Generator, procedures direct the operator to adjust the speed of the generator until the synchroscope is rotating slowly in the fast direction. The two parameters indicated by the synchroscope are:

A. Current and voltage differences.

✓B. Frequency and phase differences.

C. Frequency and current differences.

D. Current and phase differences.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]: 062 A4.03 [2.8/2.9] [41.7]

Running & incoming voltages are __not__ addressed

References: OPL271MTGC

Also low discrim value

LP/Objective: OPL271MTGC B.9

History: INPO8 155

Level: Memory

Comments: FHW 12/02 062 A4.03; modified inpo question for SQN MTGC.

are nearly equal and the electrical speeds (60 hertz) are nearly equal. The only difference would be generator frequency slightly higher than system frequency in order to have a sync. scope turning slow in the fast direction.

- When a generating unit is to be paralleled to a running power system, its voltage, speed and synchronous phase position must be balanced to those of the system. If speed or phase position is not matched, excessive current flow could occur as the generator is tied to the system. Generator damage could result from the high current flow through the windings.

- To parallel a generator to an operating system, four conditions must be satisfied:
  - Phase rotations must be the same (fixed phase connections).
  - Speeds must be the same, to ensure frequency of the alternating cycles are the same.
  - Voltage levels must be the same, to prevent reactive power flow when the generator breakers are closed.
  - Voltage phase positions must be the same (in-phase) to prevent heavy current flow through the generator windings during acceleration or deceleration of the generator speed that would be required to match system voltage.

- Speed and voltage of the incoming generator can be varied until the system frequency and voltage are matched. The revolving speed of the synchroscope is proportional to the difference in speed between the generator and the system. When the two speeds are matched, the sync' needle will be stationary or moving very slowly in the fast direction.

- The synchroscope dial indicates the angular phase difference between the running and incoming voltages. When the needle points to zero phase difference and is stationary, the two voltages are in phase and can be paralleled.

NOTE: 12:00 position is zero phase difference. Startup procedure parallels at 5 minutes prior to 12:00.

- If the incoming generator is tied-on in phase but at less than synchronous speed, it will be motored by the system to accelerate it to synchronous speed. Generator winding damage could occur if excessive current flow is required to develop the motor torque required for acceleration.

- If the incoming generator is in-phase but at a speed higher than system frequency, the generator will supply a high current flow to the system when the tie breaker is closed to decelerate the generator to synchronous speed. Generator damage could occur if excessive current flow is required to retard the speed.

- If frequency is at synchronous speed but the incoming and running voltages are OUT OF PHASE, heavy current flow will exist while the generator is accelerated or decelerated to bring its voltage to synchronous phase position. Once again, generator damage could be the consequence of the improper operation.

- When paralleling a generator to a system, frequency and phase position of the two voltages must be matched. Proper synchronizing of the two voltages will ensure that little or no transfer of energy (true or reactive power) occurs between the incoming and running equipment when the tie breaker is closed.

- Generator Capability

  - Power consumed (useful work) and dissipated (wasted work) in a pure resistive circuit depends on the generated voltage and current and the resistance of the circuit. Typical dissipated power (I^2R losses) is the heat loss from a conductor while current is flowing. Power in a resistive circuit is always positive, outgoing power which is completely consumed and dissipated in the circuit. The total power generated is the algebraic

95. ODCM-B.5 001

This question has reference material attached.

A lightning strike has disabled 0-LS-27-225, Cooling Tower Blowdown Effluent Line and causing both unit's blowdown to isolate. Repairs to the level switch will take about 45 days from today.

Unit 1 needs to restore blowdown to cooling tower blowdown today. Which ONE of the following lists the actions required to COMPLETELY address this situation?

A ... AND:

A. Have the lineup for blowdown to cooling tower blowdown double verified and have two samples from each steam generator analyzed independently. Jumper the circuitry so that the blowdown release may be initiated. Initiate blowdown to cooling tower blowdown.

B. Jumper the circuitry so that the blowdown release may be initiated. Initiate blowdown to cooling tower blowdown. If RCS dose equivalent I-131 is less than 0.01 microcuries per gram, sample the steam generators every 24 hours. Notify the Chemistry Supervisor when blowdown has been in service for 30 days without 0-LS-27-225 for inclusion in the Annual Radioactive Effluent Release Report.

C. Estimate the cooling tower blowdown flow locally. Jumper the circuitry so that the blowdown release may be initiated. Initiate blowdown to cooling tower blowdown. Estimate cooling tower blowdown flow locally every 4 hours during blowdown. No further notifications are required.

✓D. Estimate the cooling tower blowdown flow locally. Jumper the circuitry so that the blowdown release may be initiated. Initiate blowdown to cooling tower blowdown. Estimate cooling tower blowdown flow locally every 4 hours during blowdown. Notify the Chemistry Supervisor when blowdown has been in service for 30 days without 0-LS-27-225 for inclusion in the Annual Radioactive Effluent Release Report.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

2.3.11 MARGINAL
LOOK AT 035 K103 (NUREG 1122, P 3.4

95. ODCM-B.5 001

K/A[CFR]: 2.3.11 [2.7/3.2] [41.11]

References: ODCM Control 1.1.1 action b, Table 1.1-1 action 33

LP/Oblective: OPL271LRW B.3

History: Procedure bank

Level: ~~Comprehension~~ M

Comments: FHW 12/02 2.3.11 **Provide copy of ODCM R46 page 13,14,15**

1/2 CONTROLS AND SURVEILLANCE REQUIREMENTS

1/2.1 INSTRUMENTATION

1/2.1.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

CONTROLS

1.1.1 In accordance with SQN Technical Specification 6.8.4.f.1, the radioactive liquid effluent monitoring instrumentation channels shown in Table 1.1-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of ODCM Control 1.2.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology and parameters in ODCM Section 6.2.

APPLICABILITY: This requirement is applicable during all releases via these pathways.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required above, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so that it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the action shown in Table 1.1-1. Exert best effort to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability could not be corrected within 30 days.

c. The provisions of Controls 1.0.3 and 1.0.4 are not applicable. Report all deviations in the Annual Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENTS

2.1.1 Each radioactive liquid effluent monitoring channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE/SENSOR CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 2.1-1.

Table 1.1-1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
(Page 1 of 2)

	Instrument	Minimum Channels OPERABLE	Action
1.	**Gross Radioactivity Monitors Providing Automatic Termination of Release**		
	a. Liquid Radwaste Effluent Line (0-RM-90-122)	1	30
	b. Steam Generator Blowdown Effluent Line (1,2-RM-90-120A,121A)	1	31
	c. Condensate Demineralizer Effluent Line (0-RM-90-225A)	1	30
2.	**Gross Radioactivity Monitors**		
	a. Essential Raw Cooling Water Effluent Header** (0-RM-90-133A,-134A,-140A,-141A)	1	32
	b. Turbine Building Sump Effluent Line (0-RM-90-212A)	1	31
3.	**Flow Rate Measurement Devices**		
	a. Liquid Radwaste Effluent Line (0-FI-77-42)	1	33
	b. Steam Generator Blowdown Effluent Line (1,2-FI-15-44, 1,2-F-15-43)	1	33
	c. Condensate Demineralizer Effluent Line (0-FR-14-456, 0-F-14-185, 0-F-14-192)	1	33
	d. Cooling Tower Blowdown Effluent Line (0-LS-27-225)	1	33
4.	**Tank Level Indicating Devices**		
	a. Condensate Storage Tank (0-L-2-230, 0-L-2-233)	1	34
5.	**Continuous Composite Sampler and Sample Flow Monitor**		
	a. Condensate Demineralizer Regenerant Effluent Line (0-FI-14-466)	1	35

** Requires minimum of 1 Channel/Header to be OPERABLE.

Table 1.1-1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
(Page 2 of 2) TABLE NOTATION

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with ODCM Control 1.2.1.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for principal gamma emitters in accordance with SR 2.2.1.1.1 and 2.2.1.1.2.

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than or equal to 0.01 μCi/g DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than 0.01 μCi/g DOSE EQUIVALENT I-131.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for principal gamma emitters at a limit of detection of at least 5.0×10^{-7} μCi/ml.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continued provided the tank liquid level is estimated during all liquid additions to the tank.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided representative batch samples of each tank to be released are taken prior to release and composited for analysis according to Table 2.2-1, footnote g.

1. OPL271C353.4 001 APPROVED RMC' as IS contro

WHICH ONE (1) of the following will cause actual pressurizer pressure to increase?

A. Pressurizer Pressure channel fails high.

B. Steam line leak.

✓C. ~~Increase in RCS temperature.~~ Turbine Gov. valve goes closed

D. RCS Boration at 100% RTP.

A. incorrect a pressure channel failing high will not increase pressure.
B. incorrect a steam line break will reduce Tave and pressure.
C. correct an increase in RCS temperature will cause expansion of liquid and pressure increase.
D. incorrect a boration will reduce Tave.

K/A[CFR]: 027AK1.02 [2.8/3.1] {41.8 41.10}

NO PPC MALFUNCTION EXIST Better → 010 A1.06 (P3.3-7

Reference: Thermo Basic Energy Concepts-SH1-PT02Sr2.doc

LP/Objectives: OPL271C353 B.4

History: Procedure Bank

Level: Comprehension

Comments: FHW 12/02 027AK1.02

NO DISCRIMINATORY VALUE

temperatures, the specific heat of liquid water is somewhat greater than 1.0 Btu/lb_m °F.

Specific heat is used to calculate the addition of heat to a system from observed temperature changes. If the specific heat is known, the heat addition (or subtraction) may be calculated by rearranging Equation 2-15:

$$q = c_p \Delta T$$

Where:

q	=	heat transferred per unit mass (Btu/lb_m)
c_p	=	specific heat capacity (Btu/lb_m °F)
ΔT	=	temperature change (°F)

Equation 2-17

This equation assumes that c_p is an average specific heat over the range of temperature change.

The specific heat of a solid does not change appreciably with pressure or expansion, but does change slightly with temperature. The specific heat of a liquid does not change appreciably with expansion, but does change with pressure and temperature. Generally, an increase in pressure reduces in the specific heat of a liquid. Conversely, an increase in liquid temperature raises the specific heat of a liquid. Therefore, it is important not to use a single, average value of specific heat over an excessively wide range of pressure and temperature.

Direct use of specific heat should be avoided in vapor and gas calculations. The values of specific heat for substances in this phase vary widely with conditions. However, specific heat for gases undergoing certain limiting processes can be used in some applications. This will be covered in more detail in the next chapter.

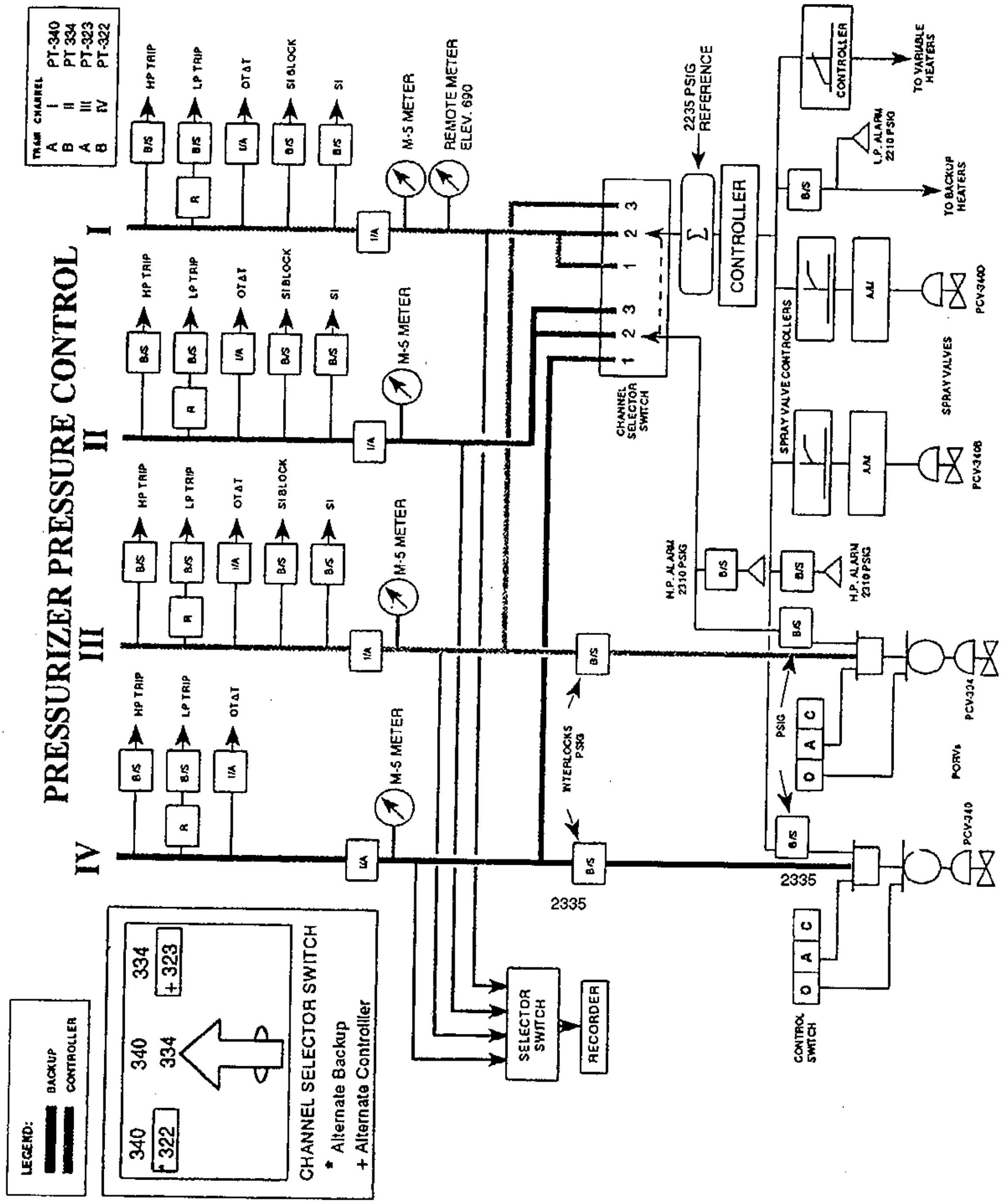
The value of specific heat for materials is tabulated in various engineering handbooks. An average value of specific heat will be given over an applicable range of temperature and pressure. It is up to the individual user to ensure that the specific heat value selected is applicable to the conditions of interest.

The most widely used specific heat is that of water at one atmosphere of pressure. The specific heat of water is used to define the British thermal unit (Btu). The Btu is the amount of heat energy that must be added to raise the temperature of one pound mass of water one degree Fahrenheit, at one atmosphere of pressure. Strictly speaking, the Btu is 1/180th of the heat needed to raise water temperature from 32°F to 212°F. It is important to note that the value of a Btu is an average taken from 32°F to 212°F. Above 212°F, the specific heat of water increases considerably, due to the large amount of heat required to change a liquid to a gas.

Most steam tables do not list values of properties for subcooled liquids. Using values for specific heat capacity, subcooled liquid property values can be calculated.

APPENDIX H

PRESSURIZER PRESSURE CONTROL



90. INPO2 231

Following a large steam line rupture, monitoring of Critical Safety Function Status Trees indicates a RED path for FR-P.1, "Pressurized Thermal Shock." Which one of the statements below correctly identifies the major component and reason for concern that Pressurized Thermal Shock conditions may result in brittle failure?

A. The RCS piping due to ~~the increased tensile stresses resulting from cooldown of unexpected severity~~. use 2nd part of "D"

B. ~~The pressurizer due to increased stresses resulting from a rapid depressurization condition at high temperature~~. The RCS piping due to... use 2nd part of "D" but change "or" to "and"

C. Th~~e steam generators due to increased stresses resulting from a rapid overpressure condition at low temperature~~. use "D" but change "or" to "and"

✓D. The reactor vessel due to increased stresses resulting from a cooldown of unexpected severity or an overpressure condition at low temperature.

a. incorrect per EPM-3-FR-P.1
b. incorrect per EPM-3-FR-P.1
c. incorrect per EPM-3-FR-P.1
d. correct per EPM-3-FR-P.1 "The thermal stress due to a rapid cooldown and the pressure stress are additive in the vessel wall."

K/A {CFR}:	040 AK1.01	[4.1/4.4]	[41.8 41.10]
References:	EPM-3-FR-P.1		
LP/Objectives:	OPL271C406 b.3		
History:	INPO2 231 Bank		
Level:	~~Comprehension~~ M		
Comments:	FHW 12/02	040 AK1.01	

INTRODUCTION

This procedure provides actions to respond to reactor vessel thermal shock, pressurized thermal shock, and low temperature overpressure conditions.

The major action categories performed in this procedure are:

Stop RCS cooldown.

Terminate ECCS if criteria satisfied.

Depressurize RCS to minimize pressure stress.

Establish normal operating conditions and stable RCS conditions.

Soak if necessary prior to further restricted cooldown.

A detailed discussion of the high level action steps, including any applicable notes or cautions, is presented on the following pages.

103. OPL271C382.4 001

~~ES-0.2, "Natural Circulation Cooldown", directs the operator to monitor RCS cooldown.~~ Which ONE of the following indicates natural circulation is in progress?

✓A. Core exit thermocouples decreasing; T-HOT decreasing; RCS subcooling increasing.

B. Core exit thermocouples decreasing; T-HOT decreasing; RCS subcooling decreasing.

C. T-HOT increasing slowly; T-COLD decreasing; RCS subcooling increasing.

D. Core exit thermocouples increasing slowly; SG pressure decreasing; RCS subcooling decreasing.

A. Correct per ES-0.2.
B. Incorrect due to subcooling decreasing.
C. Incorrect due to T-COLD decreasing.
D. Incorrect due to RCS subcooling decreasing.

K/A [CFR]: E09 EA1.1 [3.5/3.5] {41.7}

All aspects of KA not completely satisfied (p 4.5-24)

Reference: EA-68-6

LP/Objectives: OPL271C382 b.4

History: Old Bank Number PL-0904

Level: Analysis

Comments: FHW 12/02 E09 EA1.1

4.2 Verification of Natural Circulation

1. **MONITOR** the following indications of natural circulation at 15- to 20-minute intervals:

   - RCS subcooling greater than 40°F. ☐
   - S/G press stable or dropping. ☐
   - T-hot stable or dropping. ☐
   - Core exit T/C stable or dropping. ☐
   - T-cold at saturation temperature for S/G pressure. ☐

2. **DETERMINE** parameter trends between monitoring intervals and **EVALUATE** if natural circulation is occurring. ☐

3. **IF** natural circulation **NOT** verified,
   THEN
   NOTIFY ASOS. ☐

4. **GO TO** Section 4.1, step in effect. ☐

END OF TEXT

74. ES-0.2-B.3 003

Given the following plant conditions:

- All reactor coolant pumps are deenergized.
- The core exit thermocouples are reading 440°F and decreasing.
- Upper head thermocouples are reading 605°F.
- The RCS cold leg temperatures are 400°F and decreasing.
- SP → The RCS pressure is 1585 psig and slowly decreasing.
- SP → The pressurizer level has rapidly increased from 35% to 48%.

Which ONE (1) of the following is the reason for the change in pressurizer level?

A. RCS pressure has decreased to the point where the safety injection pumps have begun injecting into the RCS.

B. Anticipated response due to uneven loop cooling and flows during natural circulation.

✓C. RCS depressurization has caused saturated conditions and a steam void in the reactor vessel head area.

D. Pressurizer level swings are expected due to steam entering the pressurizer and surge line.

A. Incorrect SI pumps start injecting at 1500 psig.
B. Incorrect by design there should be no uneven loop flow.
C. Correct per steam tables. Sat temp for 1600 psia is 604.87°F.
D. Incorrect, the RCS temperature is subcooled.

K/A {CFR}: E10 EK3.1 [3.4/3.7] {41.5 41.10}

The Q Does ~~not~~ meet all the criteria for the KA

References: ES-0.2

This is a tough KA to meet!

LP/Objectives: OPL271C382, b.3

ok

History: Procedure Bank

Level: Comprehension ← weak

Comments: FHW 12/02 E10 EK3.1 **Provide steam tables**

The Q has very low discrim val

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE "Unexpected pressurizer level changes indicative of vessel voiding" is defined as an unexplained level rise when reducing RCS pressure OR an unexplained level drop when raising RCS pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	**MONITOR** for steam voids in reactor vessel:	
	a. **CHECK** for the following indications of steam voids: • unexpected pressurizer level changes indicative of voiding **OR** • RVLIS upper plenum range less than 98%.	a. **GO TO** Step 17.
	b. **REPRESSURIZE** RCS within limits of Tech Spec temperature/pressure limit curves to collapse voids in system and **CONTINUE** cooldown. **IF** repressurization is NOT desirable **OR** NOT possible, **THEN** **PERFORM** one of the following: • **GO TO** ES-0.3, Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS). **OR** • **GO TO** ES-0.4, Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS).	

113. RCP-B.12 002
Which ONE ~~(1)~~ of the following is the reason that the RCP's have an anti-reverse rotation device installed on the pump rotor?

? A. Prevent ~~reverse~~ #ANY reverse flow through a tripped RCP when other RCP's are running.

B. Prevent overheating of pump bearings due to a tripped RCP rotating without normal internal cooling flow.

C. Prevent damage to the pump thrust bearing due to operation in the reverse direction.

✓D. Prevent stator winding damage due to excessive pump starting current.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A [CFR] 003 A1.03 [2.6/2.6] [41.5] (may have to reconfigure Q a bit)
NG 003 K6.08

Reference: RCP system description section 2.0

LP/Objective: RCP System Description Objective 12.

History: System bank

Level: Memory

Comments: FHW 12/02 003 A1.03

ANS "A" is correct according to the phrase on next page

RCP Flywheel

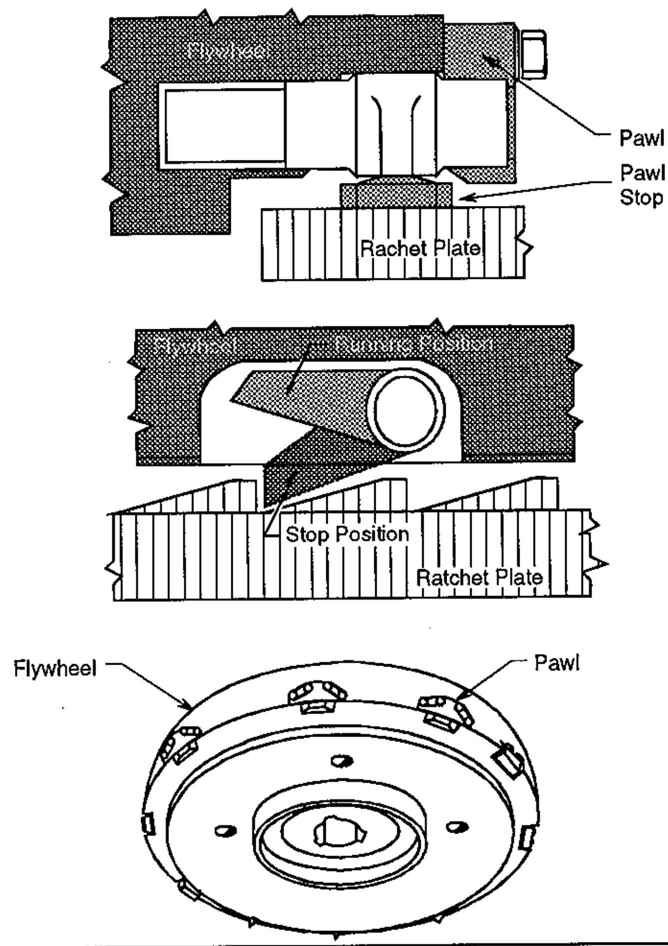
Purpose

The RCP flywheel has the following purposes:

- Ensures continuation of forced flow for a short time following a reactor trip and loss of power to the RCPs. The flywheel increases the rotational inertia of the motor so that the pump coast down time becomes longer.
- Prevents a deenergized pump from rotating backwards when another pump is running and causing reverse flow through the inactive loop. The flywheel is equipped with an anti-reverse rotation device consisting of pawls mounted on the flywheel that engage the motor frame when the pump starts to turn backwards. Starting the motor while rotating backwards would draw excessive starting currents which would overheat the motor.

Basic Block Diagram

The RCP flywheel and anti-rotation device are depicted below.



RCP-14

Continued on next page

114. RCS-TEMP-B.2 008
Which ONE ~~(1)~~ of the following temperature elements is used to supply a temperature compensation signal to correct for the Reactor Coolant liquid-density and/or steam-density changes for the Reactor Vessel Level Indicating System (RVLIS) level transmitter outputs?

A. RCS wide-range cold-leg temperature elements.

✓B. RCS wide-range hot-leg temperature elements.

C. Hot-leg RVLIS instrument-line temperature elements.

D. Cold-leg RVLIS instrument-line temperature elements.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A [CFR]: 016 A3.02 [2.9/2.9] [41.7]

Reference: 1-47W610-68-7
RCS temperature system description page 2-4.

LP/Objective: OPL271RCSTEMP B.11

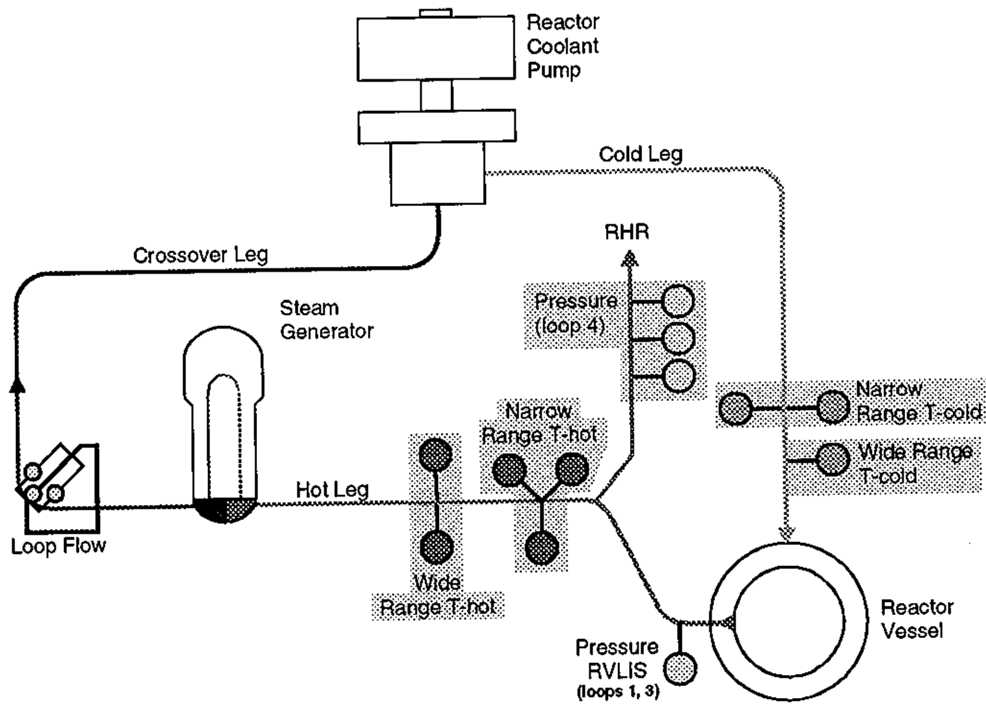
History: System bank, old Bank Number PL-1148.

Level: Memory

Comments: FHW 12/02 016 A3.02

Piping, Continued

Instruments The following drawing and tables describe instruments associated with the RCS:



RCS-04

Narrow Range Hot Leg RTDs

Instrument	Purpose	Description	Location
TE-68-2A,B,D TE-68-25A,B,D TE-68-44A,B,D TE-68-67A,B,D	Provides input into: • loop ΔT circuit • T_{avg} circuit • control and protection functions	3 RTDs, placed in thermowell, 120° apart to minimize temperature streaming effects	Each loop hot leg piping has three thermowells after the piping has exited the cavity wall and prior to the S/Gs

Continued on next page

Piping, Continued

Instruments (continued)

Narrow Range Cold Leg RTDs

Instrument	Purpose	Description	Location
TE-68-14A,B TE-68-37A,B TE-68-56A,B TE-68-79A,B	Provides input into: • loop ΔT circuit • T_{avg} circuit • control and protection functions	2 RTDs, placed in thermowell, 45° apart.	Each loop cold leg piping has two thermowells downstream of RCP discharge and prior to the piping entering the cavity wall.

Wide Range Hot Leg RTDs

Instrument	Purpose	Description	Location
TE-68-1,C TE-68-24,C TE-68-43,C TE-68-65,C	Provides input to: • indication • RVLIS • LTOP • Exosensor	2 RTDs, placed in thermowell. PAM Instrument	Each loop hot leg piping has two thermowells after the piping has exited the cavity wall and prior to the S/Gs.

Wide Range Cold Leg RTDs

Instrument	Purpose	Description	Location
TE-68-18 TE-68-41 TE-68-60 TE-68-83	Provides input to: • indication • LTOP	1 RTD, placed in thermowell PAM Instrument	Each loop cold leg piping has one thermowell downstream of RCP discharge and prior to the piping entering the cavity wall.

Continued on next page

118. REP-B.1.D 002

Select the **LOWEST** emergency classification from the list below that requires activation of the Technical Support Center. Do **NOT** consider a security event.

A. Notification of Unusual Event

✓B. Alert

C. Site Area Emergency

D. General Emergency

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference. ok ←

K/A[CFR]: 2.4.~~39~~ [3.3 - 3.1] [45.11]
2.4.~~29~~ [2.6 - 4.0] [43.2]
~~2.4.41 Betts~~

Reference: EPIP-6 R34 section 3.2.3.

LP/Objective: OPL271C198, B.1.

History: Procedure bank, old Bank Number PL-0308

Level: Memory

Comments: FHW 12/02 2.4.39

3.2 Activation of the TSC

3.2.1 Shift Manager (SM)

The SM will activate the TSC and OSC by announcing the emergency condition by one or more of the following methods.

A. Plant Public Address (PA) announcement.

B. The SM or Operations Clerk will normally activate the Emergency Paging System (EPS) or, contact the persons designated on the call list. If the EPS cannot be activated from the site, contact the Operations Duty Specialist (ODS) on the ringdown line or 5-751-1700 and have the EPS activated from the CECC.

C. The SM may activate the onsite emergency sirens at an "Alert" and shall activate the sirens at a "Site Area Emergency" or "General Emergency."

3.2.2 Call List

The Emergency Preparedness Manager (EPM) shall maintain a call list listing all TSC personnel by name, plant and home telephone numbers. The REP Call List will be updated at least quarterly by the EPM or designee with input by the appropriate section/group supervisors. The list will be provided to the SM and placed in the TSC.

3.2.3 Response

Personnel performing the following REP functions should report to the TSC, or the assigned TSC support locations (see NP-REP Appendix B Figure B-3 for TSC Layout), upon announcement of an "ALERT" or higher emergency classification or at the direction of the SED.

A. Site Vice President
B. Site Emergency Director
C. Operations Manager
D. Technical Assessment Manager
E. Operations Advisor, TAT
F. Site Security Manager
G. Radiological Control Manager (RCM)
H. Chemistry Manager
I. NRC Coordinator
J. Control Room Communicator (affected Unit Control Room)
K. EP Manager
L. TSC Clerical/Logkeeper Staff (Clerical will be called)
M. Maintenance Manager
N. Technical Assessment Team
O. Operations Communicator
P. Other Plant staff the SED determines to be necessary to support TSC functions will be called.

121. RMS-B.2 004
Which ONE ~~(X)~~ of the following Common Radiation Monitor 0-XA-55-12B annunciators in alarm indicates an area high radiation condition?

A. (A-5) 0-RA-90-132A SERVICE BLDG VENT MON HIGH RAD

B. (B-1) 0-RA-90-101A AUX BLDG VENT MONITOR HI RAD

✓C. (B-3) 0-RA-90-102A FUEL POOL RAD MONITOR HIGH RAD

D. (C-7) 0-RA-90-125A MAIN CNTRL RM INTAKE MON HIGH RAD

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A: 072 ~~A1.01~~ [3.4/3.6] [41.5]

Reference: 0-XA-55-12B (B-3)

LP/Objective: OPL271C013, B.2

Level: ~~Memory~~

History System bank (Developed 7/23/98)

Comments: FHW 12/02 072 A1.01

Source	Setpoint	0-RA-90-102A FUEL POOL RAD MONITOR HIGH RAD
SER 760 0-RE-90-102 Retransmitted to U-2 SER 2243	50 mr/hr > 1 second	

Probable Causes

1. High radiation in spent fuel pit area elevation 734.

Corrective Actions

[1] **VERIFY** the following:
  a. Auxiliary Building General Supply and Exhaust and Fuel Handling exhaust isolate (A-Train) (1-M-9).
  b. Auxiliary Building Gas Treatment System starts (1-M-9).

[2] **CHECK** 0-RM-90-102 and 0-RM-90-103 on 0-M-12 to verify alarm.

[3] **IF** high radiation alarm valid, **THEN**
  [a] **ANNOUNCE** "High Radiation at spent Fuel Pool Area" over PA system.
  [b] **NOTIFY** SM.
  [c] **NOTIFY** RADCON.

[4] **IF** B-Train ABI has not actuated from a valid High Radiation condition, **THEN**
  INITIATE manually B-Train Auxiliary Building Ventilation Isolation via **[1-HS-30-101B]** or **[2-HS-30-101B]** (M-6).

[5] **IF** fuel handling in the Spent Fuel Pit is in progress, **THEN**
  REFER TO AOP-M.04, *Refueling Malfunctions.*

[6] **REFER TO** AOP-M.06, *Loss of Spent Fuel Cooling*.

[7] **IF** Auxiliary Building Ventilation Isolation resulted from an invalid ABI signal, **THEN** ,
  REFER to 0-SO-30-10 *Auxiliary Building Ventilation Systems* to recover from ABI.

[8] **EVALUATE** Technical Specifications 3.3.3.1 and 3.9.12.

[9] **INITIATE** Corrective Actions.

[10] **WHEN** conditions return to normal, **THEN**
  RETURN Auxiliary Building Ventilation System to normal in accordance with 0-SO-30-10, *Auxiliary Building Ventilation Systems.*

References

45B655-12B-0,
47W610-90-1

122. RMS-B.9 001
A Waste Gas Decay Tank is being vented to atmosphere using the normal gaseous waste discharge path. If high gaseous activity exists in this tank, which ONE ~~(1)~~ of the following radiation monitors would generate an alarm?

A. Auxiliary Building Ventilation Monitor [RE-90-101].

B. Containment Purge Air Exhaust Monitor [RE-90-130].

C. Service Building Ventilation Monitor [RE-90-132].

✓D. Shield Building Ventilation Monitor [RE-90-400].

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A {CFR}: ~~071 A1.06 [2.5/2.8] [41.5]~~ Not so good
~~073 K1.01 (3.6 - 3.9)~~ " " "
~~073 A4.01 (3.9 - 3.9)~~ " " "
Better = 071 A4.09

Reference: 1,2-47W611-77-4
0-SO-77-15

LP/Objective: OPL271GRW B.11

History: System bank

Level: Memory

Comments: FHW 12/02 071 A1.06

Date ___________

6.0 NORMAL OPERATION (Continued)

NOTE Radiation Control Valve 0-RCV-77-119 is interlocked with 1-FT-30-150 and 2-FT-30-165, and will terminate the release if the fan stops running.

[9] **VERIFY** the applicable ABGTS fan is running. ________ UO

[10] **IF** [0-RE-90-118] is operable, **THEN**

ENSURE 0-RM-90-118 is in service. ________ UO

[11] **IF** [0-RE-90-118] is INOPERABLE, **THEN**

ENSURE Compliance with actions of ODCM 1.1.2, **AND** WO initiated for corrective action. ________ UO

NOTE If the selected unit shield bldg. radiation monitor is inoperable, then the following step may be **N/A**.

[12] **VERIFY** radiation monitor RM-90-400 is in service and operable for the Shield Building vent indicated in step **[3]**. ________ UO

[13] **IF** the shield bldg vent release path selected in step **[3]**, radiation monitor RM-90-400 is inoperable, **THEN**

NOTIFY the U-1 SRO to take appropriate actions in accordance with Section 1.1.2 of the ODCM, **AND** WO initiated for corrective action. ________ UO

43. AOP-P.02-B.4 002

A loss of 125V DC Vital Battery Board I occurs during a surveillance test on 1A-A diesel generator. The diesel generator has been paralleled to the 1B 6.9kV Unit Board and is sharing the load. The loss of 125V DC Vital Battery Board I will result in which ONE (1) of the following?

A. diesel trip due to underfrequency.

B. loss of control to the diesel generator output breaker ONLY.

✓C. loss of speed and voltage control from the control room along with the ability to shutdown the diesel from the control room.

D. no effect - 125V DC Vital Battery Board I power only affects the ability to start the diesel.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A {CFR}:	✓058 AK3.01	[3.4/3.7]	41.5, 41.10
Better →	063 K3.02	[3.5/3.7]	41.7
OR Better →	058 AA2.03	[3.5/3.9]	43.5

References: 45N767-5

LP/Objectives: OPL271C364, b.4

History: Procedure Bank

Level: ~~Comprehension~~ mem

Comments: FHW 12/02 058 AK3.01 (LOAD memory Q)

123. RPS-B.5.B 001

The Reactor Protection System manual handswitches in the Control Room are operated in an attempt to trip the Reactor during an ATWS event.

Which ONE ~~(s)~~ of the following describes how the Reactor Trip Breaker's trip attachments should respond to operation of a manual Reactor Trip handswitch?

A. The SSPS undervoltage trip attachments and the 125V-DC trip attachments must both deenergize to trip the Reactor.

✓B. The SSPS undervoltage trip coils must deenergize, while the 125V-DC trip coils must energize to trip the Reactor.

Not plausible

C. The SSPS undervoltage coils will ~~deenergize~~ energized and the 125V-DC trip coils are ~~unaffected.~~ deenergized

D. The 125V-DC trip coils will energize, while the SSPS undervoltage coils will be unaffected.

A. Incorrect, 125V-DC trip coil will energize.
B. Correct, per reference.
C. Incorrect, 125V-DC trip coil will energize.
D. Incorrect, the undervoltage coil will deenergize.

K/A[CFR]: 012 A3.07 (4.0 / 4.0)
029 EK2.06 [2.9/3.1] {41.7} Either one is ok

Reference: 0-47W611-99-1

LP/Objective: OPL271RPS, b.9

History: System Bank old Bank Number PL-0811

Level: Comprehension

Comments: FHW 12/02 029 EK2.06

94. MFW 001

The Main Feedwater Regulating Bypass Valves receive level input signals from:

A. Associated SG's Motor Driven Auxiliary Feedwater Level transmitters.

✓B. Associated SG's Turbine Driven Auxiliary Feedwater Level transmitters. INSUFFICIENT TECH INFO TO CONFIRM

C. Narrow range level signal from the median selector on each SG control circuit. I think need DWG 47W611-3-

D. The non-PAM narrow range level transmitter from the associated SG.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

KA[CFR]: 059 K1.04 [3.4/3.4] [41.2-9] ok

References: 1,2-47W611-3-2

LP/Objective: Steam Generator System description section 2. Objective 2

History: System bank

Level: Memory

Comments: FHW 12/02 059 K1.04

1. FW-B.5 001

Given the following plant conditions and information:

- Unit is at 100% rated thermal power
- Feedwater Master Controller and Feedwater Pump Speed Controllers are in AUTOMATIC.
- Main Feedwater Regulating Valves are in AUTOMATIC.
- All four main feedwater flows start increasing with level in all four steam generators trending upwards.

For information:

- PT-1-33 is a Main steam header pressure transmitter
- PT-3-1 is a Main feedwater header pressure transmitter

Which ONE (1) of the following describes the instrument failures that could have caused this transient?

A. PT-1-33 has failed LOW and PT-3-1 has failed LOW.

✓B. PT-1-33 has failed HIGH and PT-3-1 has failed LOW.

C. PT-1-33 has failed HIGH and PT-3-1 has failed HIGH.

D. PT-1-33 has failed LOW and PT-3-1 has failed HIGH.

1. FW-B.5 001

A. PT-1-33 failing low would cause a high delta-P and the MFPT control system would reduce MFPT speed causing low feed flow and decreasing SG levels. The results of PT-3-1 failing low are described in "B" below.

B. Correct. Feedwater header pressure is normally higher than steam header pressure by a programmed value of 80 psid at 0% totalized steam flow and 195 psid at 100% totalized steam flow. PT-1-33 failing high OR PT-3-1 failing low would indicate a lower than normal delta-P between steam header pressure and FW header pressure. This would cause the MFPT control system to increase speed in an attempt to restore programmed delta-P. This increased delta-P would increase FW flow and cause SG level to trend upward. The FW regulating valves would be closing in an attempt to reduce SG level back to programmed value.

C. The results of PT-1-33 failing high are described in "B" above. PT-3-1 failing high would cause a high delta-P and the MFPT control system would reduce MFPT speed causing low feed flow and decreasing SG levels.

D. PT-1-33 failing low would cause a high delta-P and the MFPT control system would reduce MFPT speed causing low feed flow and decreasing SG levels. PT-3-1 failing high would cause a high delta-P and the MFPT control system would reduce MFPT speed causing low feed flow and decreasing SG levels.

K/A{CFR}: 039 K3.04 [2.5/2.6] [41.7]
059 K1.04 [3.4/3.4] {41.2, 41.3, 41.4, 41.5, 41.6, 41.7, 41.8, 41.9}
059 K6.03 [1.9/2.1] {41.7}
059 K6.09 [2.4/2.6] {41.7}
059 A2.11 [3.0/3.3] {41.5, 43.5}

References: 47W611-3-2
OPL271CCOND/FW

LP/Objectives: OPL271C034 Obj. B.5

History: WBNOPS--1.bnk Q# SYS003D.14 001(Modified stem)

Level: Analysis

Comments: LP-5/2000;. FW 004; FHW 12/02 039 K3.04

Intermediate Pressure Feedwater Heaters

- Condensate from the intermediate pressure heater strings is then routed to the main feed pumps

Main feed pumps

- Pumps: ~65% capacity each; 20,000 gpm at 1680' TDH

- Pump controls
  - Handswitch located on M-3
  - RPM, suction flow,

- Speed control

The main feed pumps maintain a programmed ΔP across the #1 heaters and the S/Gs.

Program setpoints: 80 psid at no load, to 195 psid at full load. Load index is totalized steam flow. The ΔP is held level @ 80 psid from 0-20% load, then is linear from 20% to 100% load.

- Controllers: discuss startup with one in auto using the master controller.

- One for both (master controller)

- This controller is susceptible to Reset Windup.
  SOER 94-001 describes reset windup in the following manner:
  > "Reset Windup" occurs when there is an error between the process signal and the setpoint of an controller for an extended period of time. During this time Integration (reset) causes the output to continue to change until high or low output saturation is reached. As a result, when the process changes such that the controller is driven back out of saturation, there may be a significant time delay before the controller output actually comes out of saturation and begins to change in the right direction. This time delay can be several minutes. When a "Reset Windup" condition is observed , the operator should take manual control of the controller , make required adjustments and place the controller back in auto when the condition has cleared.

Automatic Pump Trip: Excessive thrust - Turbine 5 mils (forward + reverse);
- Pump 5 mils (forward + reverse);
Low bearing oil pressure 10 psig;
Electrical Fault.;
Suction valve not fully open;
Low injection water pressure;
Low condenser vacuum (feed inlet and outlet valves closed will cause this);
Feedwater isolation signal;
Overspeed (mechanical):
Manual.

NOTE: High Vibration: vibration trips on both the pump and turbine have been disconnected These are shown on the electrical prints and referenced to the drawing notes indicating such.

111. RADWASTE-B.12 008
What TWO conditions will independently cause automatic closure of Liquid Radwaste Release Valve, 0-RCV-77-43? collectively Drawing shows an "and" gate at (F, 9.5)

✓A. Low cooling tower blowdown flow, and high radiation sensed in the release header.

B. Low cooling tower blowdown flow, and high radiation sensed in the cooling tower blowdown flow.

C. High release header flow, and high radiation sensed in the release header.

D. High release header flow, and high radiation sensed in the cooling tower blowdown flow.

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A {CFR}:	068 A3.02 [3.6/3.6] [41.7]
References:	1,2-47W611-77-2
LP/Objectives:	OPL271LRW B.12
History:	System bank
Level:	Memory
Comments:	Reviewed by T Jetton, FHW 12/02 068 A3.02

86. FW-B.5.B 004

A ~~Main Feedwater~~ (MFW) Pump is provided ~~with various~~ protection devices that will trip it when certain parameters ~~are reached. The SSPS provides some inputs for that protection.~~

Which ONE ~~(1)~~ of the following describes the SSPS input to the MFW Pump trip circuitry?

A. The SSPS provides a signal to a 120v AC trip solenoid valve which dumps trip oil thus tripping the pump.

B. The SSPS provides a signal to the overspeed trip plunger which dumps trip oil thus tripping the pump.

C. The SSPS provides a signal to a 48v DC trip solenoid valve which dumps trip oil thus tripping the pump.

✓D. The SSPS provides signals to two 125v DC trip solenoid valves which dump trip oil thus tripping the pump. not corr on dwg.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

KA[CFR]:	063 K4.04 [2.6/2.9] [41.7] ok 059 K4.16 (3.1-3.2) Better
References:	1,2-47W611-99-4
LP/Objective:	OPL271COND.FW B.12
History:	System bank; 9/2/97 Makeup Audit Exam.
Level:	Memory
Comments:	FHW 12/02 063 K4.04

115. RCW-B.9 001
Which ONE (1) of the following is the source of water for the Raw Cooling Water System?

ASSUME RCW ≠ SWS

A. Emergency Raw Cooling Water System

B. Raw Service Water System

✓C. Condenser Circulating Water System

D. High Pressure Fire Protection System

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference. Fire protection is used for a backup to the CTLP bearing lube water.

K/A[CFR]:	075 K1.01 [2.5/2.5] [41.2-9]
References:	1-47W844-1
LP/Objective:	OPL271CCW B.9
History:	System bank
Level:	Memory
Comments:	FHW 12/02 075 K1.01

25. SDCS-B.12 023
Unit One is operating at 100% power ~~when a leak develops near the condenser vacuum breaker. The C-9 interlock~~ window is dark.

Which ONE (1) of the following conditions is required for operation of the Steam Dumps?

✓A. The Steam Dump System will be enabled only after Main Condenser pressure is 3.4 psia or less and one (or more) Condenser Circulating Water Pump is running.

B. The Steam Dump System is ~~still~~ always available for operation at power

C. The Steam Dump System will be enabled only after Main Condenser pressure is 2.7 psia or less and one (or more) Condenser Circulating Water Pump is running.

D. The Steam Dump System will be enabled only after Main Condenser pressure is 1.72 psia and one (or more) Condenser Circulating Water Pump is running.

A. Correct per 1-AR-M4-A (E-6), 6 to 7 inches Hg abs or approx. 3.4 psia.
B. Incorrect C-9 is lit when available, see 1-AR-M4-A (E-6).
C. Incorrect outside PS setpoint.
D. Incorrect outside PS setpoint, this value is the manual trip setpoint per AOP-S.02.

Student must understand when the C-9 interlock window is lit the condenser is available. How can "A" be true if C-9 window is dark?

K/A[CFR]: → OK 041 A4.08 (3.0 - 3.1)
→ NO 051 AK3.01 [2.8/3.1] {41.5 41.10} Therefore delete Q as recommended.

Reference: 1-AR-M4-A (E-6)
TI-28

LP/Objective: OPL271C030, b.10

History: System bank, old Bank Number PL-1440

Level: ~~Comprehension~~ Mem

Comment: FHW 12/02 051 AK3.01

C-9 CONDENSER INTERLOCK

(Normally Lit at Full Power)

Source

SER 1934
Condenser vacuum PS-2-1B and PS-2-7C
CCWP "A"-6.9 UNIT Bd "A", Comp 9
CCWP "B"-6.9 UNIT Bd. "B", Comp 5
CCWP "C"-6.9 UNIT Bd "D", Comp 9

Setpoint

2/2 PS 6 to 7" Hg abs (approx. 3.4 psia)
AND
one condenser circulating water pump breaker closed.

Probable Causes

1. Main condenser has sufficient vacuum to support steam dump operation with $\geq$ one CCWP running.

NOTE The presence of this alarm allows steam dumps to operate.

Corrective Actions

[1] **VERIFY** condenser pressure less than 3.4 psia.

[2] **VERIFY** at least one condenser circulating water pump operation.

[3] **IF** C-9 goes dark during normal operation, **THEN** **REFER** to AOP-S.02, *Loss of Condenser Vacuum*.

References

45N601-1,
45B655-04A-0,
47W611-2

CONTROLS

TAVG

547 °F - 578.2 °F
(0 - 100% power)

PZR LEVEL

24.7% - 60%
(547 °F - 578.2 °F T_{avg} [Auct. Hi T_{avg}])

FEEDPUMP SPEED CONTROL

80 - 195 psid, (0 - 20 %, constant 80 psig)
(20 - 100% total steam flow)

S/G LEVEL

33 - 44% (0 - 20% turbine load)
44% (20 - 100% turbine load)

ROD CONTROL

Auto: 8 spm; 1.5°F → 3.0°F error
8 spm - 72 spm; 3°F → 5°F error

Manual: 48 spm

Bank Select: 48 spm control rods
64 spm shutdown rods

FP Rod Insertion Limit: 182 steps on "D" Bank

FIRST-STAGE IMPULSE PRESSURE

0 - 628 psia
0 - 100% power

STEAM DUMPS

Blocking:

Steam Dump Bypass Interlock Switches (M-4) in OFF

Condenser not available (absence of C-9)

Lo-Lo T_{avg} (P-12) locks out all steam dumps; interlock can be bypassed for 3 cooldown valves using the two bypass interlock switches on M-4.

Arming:

Load Rejection: 10% load decrease in a 2 minute time constant as sensed by PT-1-72 (C-7).

Reactor Trip: P-4 from 'A' Tr. Rx Trip Breakers ('B' Tr. P-4 places Rx Trip Controller I/S.)

Mode Selector Switch (M-4) in STEAM PRESSURE

Opening:

Tave Mode ⇒

Load Reject: Tave - Tref (PT-1-73)
(2°F → 18°F = 0 → 100% open)
Trip Open - ½ @ 10°F; ½ @ 18°F

Reactor Trip: Tave - 552°F (Fixed Reference Signal)
(0°F → 50°F = 0 → 100% open)
Trip Open - ½ @ 25°F; ½ @ 50°F

Pressure Mode: Auto = steam pressure - setpoint
Manual = Operator Controlled

120. RHR-B.13.H 001
Given the following plant conditions:

- Plant cooldown in progress using two trains of RHR.
- The flow is 2500 gpm per train.
- The RCS cooldown rate is too high.

Which ONE (1) of the following operator actions is required to **DECREASE** the RCS cooldown rate while maintaining constant RHR flow?

A. ~~Throttle~~ ~~open~~ closed the Component Cooling water outlet valves on the RHR heat exchangers to slow the cooldown rate. NOT plausible if "open"

B. Throttle open the RHR heat exchanger bypass valve.

C. Throttle the outlet valves on the RHR heat exchangers closed to decrease the cooldown rate.

✓D. Throttle open the RHR heat exchanger bypass valve and throttle closed the RHR heat exchanger outlet valves.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference and decreased flow could lead to vortex.
D. Correct per reference.

K/A [CFR]: 005 A1.01 [3.5/3.6] [41.5]

Reference: 0-SO-74-1 R39 section 5.6

LP/Objective: OPL271RHR B.12

History: Systems bank

Level: Memory

Comments: FHW 12/02 005 A1.01

Unit ____________ Date__________

5.6 Startup of RHR System for Normal Cooldown Mode (Continued)

CAUTION **Flow through FCV-74-16, FCV-74-28, and FCV-74-32 must be coordinated to achieve RCS cooldown rate of 50°F per hour (100°F per hour absolute maximum).**

NOTE 1 The valve for the RHR train in service may be throttled to obtain desired cooling rate.

NOTE 2 Steps **[29]** and **[30]** should be performed concurrently.

[29] **SLOWLY THROTTLE** [FCV-74-32] RHR Hx bypass to avoid thermal shock. ________

[30] **THROTTLE** the RHR Hx valve for the RHR train being placed in service. (**N/A** other valve)

VALVE NO.	FUNCTION	INITIALS
FCV-74-16	RHR Hx A Outlet	________
FCV-74-28	RHR Hx B Outlet	________

CAUTION **RHR must be capable of 4 Loop Cold Leg injection while in Mode 4.**

[31] **IF** required, **THEN**

CLOSE the RHR cross-tie valve for the train **not** in service (**N/A** other valve):

VALVE NO.	FUNCTION	INITIALS
FCV-74-33	RHR Hx A Outlet	________
FCV-74-35	RHR Hx B Outlet	________

[32] **IF** two TRAIN RHR cooldown is desired, **THEN**

GO TO Section **6.2**. ________

END OF TEXT

RO/SRO/KA Cross Reference

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
001 AK1.03	AOP-C.01-B.1 002	B	C	A
001 K.301	001 K.301	B	A	N
001 K5.30	CONTROL*RODS 020	B	M	A
003 A1.03	RCP-B.12 002	B	M	A
004 A3.07	004 A3.07	B	C	N
005 A1.01	RHR-B.13.H 001	B	M	N
006 A1.07	ECCS-B.3 002	B	M	A
006 K6.19	006 K6.19	B	C	N
008 K4.02	CCS-B.9.A 001	B	A	A
009 EK1.01	INPO3 955	B	C	A
010 A2.02	010 A2.02	B	M	N
012 K6.07	012 K6.07	B	C	N
013 K4.01	013 K4.01	B	M	N
015 AK2.08	AOP-R.05-B.5 001	B	M	A
016 A3.02	RCS-TEMP-B.2 008	B	M	A
016 K1.01	INCORE-B.1.B 002	B	M	A
017 K6.01	INCORE-B.1.D 003	B	M	A
022 K2.01	022 K2.01	B	C	N
024 AA1.26	ES-0.1-B.1 002	B	C	A
025 AK2.05	FR-Z.1-B.2 001	B	C	A
025 K5.02	CTMT-B.11 001	B	C	A
027 AK1.02	OPL271C353.4 001	B	C	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
029 EK2.06	RPS-B.5.B 001	B	C	A
034 A1.02	034 A1.02	B	A	N
034 K4.02	FH-B.12.B 001	B	M	A
035 K5.01	035 K5.01	B	C	N
037 2.1.7	AOP-R.01-B.2 003	B	A	A
037 AA1.11	AOP-R.01-B.2 004	B	M	A
038 EK3.08	INPO2 976	B	C	A
039 K3.04	FW-B.5 001	B	A	A
040 AK1.01	INPO2 231	B	C	A
051 AK3.01	SDCS-B.12 023	B	C	A
058 AK3.01	AOP-P.02-B.4 002	B	C	A
059 A4.01	059 A4.01	B	C	N
059 K1.04	MFW 001	B	M	A
060 AK2.02	060 AK2.02	B	C	N
061 AK2.01	061 AK2.01	B	A	N
061 K2.01	061 K2.01	B	M	N
061 K6.01	061 K6.01	B	M	N
062 A4.03	INPO8 155	B	M	A
063 A2.01	063 A2.01	B	M	N
063 K4.04	FW-B.5.B 004	B	M	A
064 K2.02	064 K2.02	B	C	N
068 A3.02	RADWASTE-B.12 008	B	M	A
069 AA1.03	CTMT-B.5 004	B	C	A
071 A1.06	RMS-B.9 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
072 A1.01	RMS-B.2 004	B	M	A
074 EK1.01	FR-C.1-B.2 003	B	C	A
074 EK2.02	FR-C.1-B.2 011	B	A	A
075 K1.01	RCW-B.9 001	B	M	A
076 AK3.05	076 AK3.05	B	C	N
078 K1.01	AIR-B.12 002	B	C	A
079 K4.01	AIR-B.5 014	B	C	A
103 A3.01	103 A3.01	B	M	N
2.1.24	2.1.24	B	A	N
2.2.11	CTMT-B.11 004	B	M	A
2.2.4	2.2.4	B	A	N
2.3.11	ODCM-B.5 001	B	C	A
2.3.9	CTMT PURGE-B.4 001	B	M	A
2.4.39	REP-B.1.D 002	B	M	A
2.4.4	AOP-R.02-B.2 001	B	C	A
E01 EA1.1	ES-0.0-B.3 001	B	C	A
E03 EA1.2	ES-1.2-B.2 006	B	C	A
E03 EK1.3	ES-1.2-B.2 004	B	M	A
E04 EK2.2	ECA-1.2-B.1 002	B	M	A
E05 EA1.3	FR-H.1-B.3 004	B	C	A
E08 EK3.2	FR-P.1 001	B	C	A
E09 EA1.1	OPL271C382.4 001	B	A	A
E10 EK3.1	ES-0.2-B.3 003	B	C	A
E14 EK1.3	ECCS-B.2 002	B	M	A
E15 EK1.1	FR-Z.2-B.2 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
003 K4.02	T/S0304.02 003	R	C	A
005 AA2.01	005 AA2.01	R	C	N
008 A2.04	008 A2.04	R	C	N
012 K2.01	RDCNT-B.7.B 001	R	C	A
015 A2.03	FISSION*PROD*POISON 052	R	M	A
022 2.4.27	022 2.4.27	R	A	N
025 2.2.13	025 2.2.13	R	M	N
029 A1.02	PI-B.10 001	R	M	A
039 K5.05	RVINT-B.8 001	R	M	A
056 A2.04	056 A2.04	R	C	N
056 AK3.01	OPL271C368.3 001	R	M	A
056 K1.03	056 K1.03	R	M	N
059 AA2.05	059 AA2.05	R	M	N
064 A3.06	D/G-B.10 002	R	A	A
067 AA2.17	AOP-C.04 001	R	M	A
068 AK2.07	D/G-B.6 012	R	M	A
068 K4.01	RADWASTE-B.12 006	R	M	A
071 A4.26	WGDS-B.12 001	R	C	A
078 K3.03	078 K3.03	R	C	N
079 2.1.1	079 2.1.1	R	M	N
086 K3.01	086 K3.01	R	C	N
2.1.16	2.1.16	R	M	N
2.1.31	PZR PRESS-B.9 006	R	C	A
2.2.26	REFUELING-B.1.G 001	R	M	A
2.3.10	2.3.10	R	C	N

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.4.20	AOP-M.01-B.3 002	R	M	A
2.4.5	EPM-4-B.7 001	R	C	A
E01 2.2.25	E01 2.2.25	R	C	N
E15 EK2.2	E15 EK2.2	R	C	N
003 DROP ROD 2.4.4	AOP-C.01-B.5 011	S	M	A
004 2.4.1	OPL271C367.1 002	S	C	A
007 2.1.10	PRT-B.7 004	S	M	A
010 2.4.47	AOP-I.04-B.2 001	S	A	A
011 2.1.6	OPL271C367.1 003	S	C	A
013 2.4.47	E-0-B.6 003	S	M	A
015 AA2.07	AOP-R.04-B.5 001	S	A	A
022 AA2.01	AOP-R.05-B.2 001	S	A	A
025 AA2.06	RHR-B.12.A 003	S	M	A
027 AA2.04	T.S-2.1-B.2 001	S	A	A
040 AA2.01	OPL271C379.3 001	S	A	A
056 AA2.18	E-0-B.3.A 007	S	C	A
069 2.1.14	069 2.1.14	S	C	N
2.1.10	OPL271C458.1 001	S	M	A
2.1.22	GO-2-B.1 003	S	M	A
2.1.7	PZR LEVEL-B.14 003	S	A	A
2.2.25	OPL271C180.3 001	S	M	A
2.2.9	SPP-9.5 001	S	M	A
2.3.1	RCI-15 001	S	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.3.2	RADIATION 002	S	M	A
2.3.3	AOP-C.04-B.5 008	S	M	A
2.4.33	OPDP-4 002	S	M	A
2.4.45	AOP-M.03-B.1 002	S	C	A
E01 EA2.1	ES-0.0-B.5 002	S	C	A
E04 EA2.1	ECA-1.2-B.2 001	S	C	A
E09 EA2.1	ES-0.2-B.3 001	S	C	A
E12 2.4.16	ECA-2.1-B.1 004	S	A	A
E15 EA2.1	E15 EA2.1	S	M	N
E16 2.4.41	E16 2.4.41	S	C	N

3. 005 AA2.01 001

Rod Cluster Control Assembly F8 is greater than 12 steps from it's group step counter demand position. Tech Spec actions for continued power operation have been met except for verifing F_Q (z) and $F^N_{\Delta H}$ are within their limits.

Which one of the following will Reactor Engineering use for evaluation of F_Q (z) and $F^N_{\Delta H}$ per Tech Specs for this situation?

A. delta I indications.

✓B. incore detector flux map.

C. core exit thermocouples.

D. NIS indications.

A. Incorrect, delta I does not determine rod power distribution.
B. Correct method to determine rod power distribution.
C. Incorrect, core exit T/Cs do not determine rod power distribution.
D. Incorrect, NIS does not determine rod power distribution.

K/A[CFR]: 005 AA2.01 [3.3/4.1] [43.5]

Reference: TS 3.1.3.1 action c.3.c).

LP/Objective: OPL271C071 B.6

History: New question.

Level: Comprehension.

Comments: FHW 12/02 005 AA2.01✓It is doubtful that a student can recall an action method in the second sub-paragraph of an LCO, therefore he must understand that the only method of obtaining the heat flux along the "z" direction of a fuel rod is by an incore probe.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours. |R219

b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours. |R219

c. With one full length rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either: |R219

   1. The rod is restored within the above alignment requirements, or |R219

   2. The remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limit of specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or |R21° |R1.

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

ACTION: (Continued)

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours. FP

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be trippable by verifying rod freedom of movement by movement of ≥ 10 steps in either direction at least once per 92 days. R219

5. 008 A2.04 001

"0-RA-90-123A CCS Liq Eff Mon High Rad" alarm is lit.

Which one of the following is the correct cause and action for this condition?

A. A leak in the Containment Spray Heat Exchanger and ensure ~~the CCS surge tank~~ makeup to surge tank vents are closed.

B. The emergency supply valve to the CCS surge tank is not fully seated and dispatch operations personnel to ensure the valve is closed and surge tank vents are closed. ID

✓C. A leak in the RHR heat exchanger and ensure the CCS surge tank vents are closed.

D. A leak exists in the CCP gear drive reservoir heat exchanger a~~nd ensure the CCS~~ proper operation of auto makeup surge tank vents are closed.

A. Incorrect, the cooling medium is ERCW.
B. Incorrect, the make up supply is ERCW and the supply has double isolation valves with a spool piece.
C. Correct per reference.
D. Incorrect, the cooling medium is ERCW.

K/A[CFR]:	008 A2.04 [3.3/3.5] [41.5 43.5]
Reference:	0-AR-M12-B (C-5) 1,2-47W845-2 2-47W845-4
LP/Objective:	OPL271CCS B.9
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 008 A2.04

4th "CCS surge tank vents are closed."

"use procedures to correct ... control..." NOT met since no alternate choice available.

Source

SER 769
0-RE-90-123
monitors outlet CCS Hx
0B1 and 0B2
Retransmitted to U-2
SER 2248

Setpoint

Refer to TI-18 for setpoint calculation

0-RA-90-123A **CCS LIQ EFF MON** **HIGH RAD**

Probable Causes

1. RHR to CCS leakage in RHR Heat Exchanger 1B or 2B. **[C.1]**
2. Performing test on monitor.

Corrective Actions

[1] **CHECK** instruments on 0-M-12 (0-RM-90-123 and 0-RR-90-123) to determine if alarm is due to high radiation. **[C.1]**

[2] IF alarm is due to high radiation, **THEN**

[a] **ENSURE** 1-FCV-70-66 and 2-FCV-70-66 Component Cooling Water Surge Tank Vents are **CLOSED**. **[C.1]**

[b] **REFER TO** AOP-R.05, RCS Leak and Leak Source Identification to locate, **AND**

ISOLATE the source of the leak. **[C.1]**

[c] **NOTIFY** Radiochemical Laboratory to sample component cooling water for radioactivity. **[C.1]**

[d] **NOTIFY** RADCON to survey for radiological hazards. **[C.1]**

[e] **CONSIDER** system "feed and bleed" in accordance with 0-SO-70-1, *Component Cooling Water System "B" Train.* **[C.1]**

References

45B655-12B-0,
47B601-90-34,
47W610-90-2

9. 022 2.4.27 001

Unit One is in mode 6 with fuel movement in progress and outside air being used for containment cooling. An outside spontaneous fire ignited near the Unit One Auxiliary Building air intake. The outside AUO reports there is a lot of smoke.

Which one of the following will prevent smoke from getting into Unit One containment?

A. Containment Purge air fans will automatically shut down.

B. Unit One Auxiliary Building supply fans will automatically shut down.

✓A C. Manual Containment vent isolation is required ~~will close the containment dampers.~~

D. Upper and Lower compartment coolers have HEPA filters.

This would be true regardless of other choices

A. Incorrect, even though the purge fans have a suction source in the auxiliary building air intake plenum, the fans will not automatically shut down, reference print 1,2-47W611-30-1.

B. Incorrect, smoke detection in the auxiliary building supply will automatically trip the auxiliary building supply fans, but the purge fans take suction from the intake plenum just as the auxiliary building supply fans, therefore smoke can still be drawn through the air intake plenum, see reference print 1,2-47W611-30-5. 1,2-47W866-2.

C. Correct, even though the containment vent isolation will close the containment dampers as well as stop the purge fans; a smoke detection signal is not part of the logic, so a manual action is required, see reference print 1,2-47W611-30-1.

D. Incorrect, the upper and lower compartment coolers' suction are located inside containment and have cooling coils, see reference 1-47W866-1.

K/A[CFR]: 022 2.4.27 [3.0/3.5] [41.10 43.5]

Reference: 0-SO-30-3
1-47W866-1
1,2-47W866-2
1 ,2-47W611-30-1
1,2-47W611-30-5
TS 3.9.9.

K/A: Knowledge of fire in plant procedures

LP/Objective: OPL217C426 B.3

History: New question.

Level: Analysis

Comments: FHW 12/02 022 2.4.27

Unit____________ Date____________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY)

CAUTION **This procedure may only be performed in MODES 5 or 6.**

[1] **INITIATE** 0-SI-CEM-30-410.2, unless not required during Mode 5 & 6, as determined by the Unit Supervisor/SRO and RCL Supervision. ________

[2] **VERIFY** there is not a containment vent isolation signal. ________

CAUTION **During MODE 6 operation with the wafer valve open, containment purge startup may result in over flow of the SFP or reactor cavity due to pressure changes in the containment building. [C.2]**

[3] **IF** in MODE 6 with the wafer valve open, **THEN**

[a] **COMPARE** SFP level to reactor cavity level. **[C.2]** ________

[b] **MONITOR** locally SFP level or reactor cavity level during purge startup. **[C.2]** ________

[4] **CHECK** Containment Purge Train A filter assembly drain loop seal sight glass level equal to or greater than 50% (AB, El 669, Penetration Room). ________

[5] **IF** the loop seal sight glass level is not equal to or greater than 50%, **THEN**

FILL loop seal in accordance with 0-SO-30-3, Section 8.5. ________

NOTE If Rad Monitor 400 is inoperable, it may be **N/A** provided the ODCM requirements are met.

[6] **ENSURE** the Shield Building Vent Monitor RM-90-400A, B, C radiation monitor is in service. ________

Unit__________ Date__________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

[7] **ENSURE** Containment Purge Exhaust monitor RM-90-130 and/or RM-90-131 operable and aligned to the train of purge fans being used in accordance with 1(2)-SO-90-2. ________

[8] **ENSURE** containment pressure between -0.1 and 0.23 psid. ________

NOTE Opening damper **[FCV-30-2]** last ensures a positive pressure will be maintained in the upper compartment with respect to the lower compartment during the startup transient, and will ensure that the lower ice doors remain closed during startup when out-of-balance flow situation may exist.

[9] **VERIFY** Power Checklist 1(2)-30-3.01 complete. ________

[10] **VERIFY** Valve Checklist 1(2)-30-3.02 complete. ________

[11] **IF** Ice condenser doors are NOT blocked closed, **THEN** **MONITOR** the Ice Condenser Lower Inlet Door Open alarm on M-6. ________

[12] **IF** Ice Condenser Lower Inlet Door Open alarm is inoperable, **THEN**
MONITOR the ice condenser door status lights on M-10 once per hour. ________

[13] **VERIFY** all containment purge fans **OFF**. ________

[14] **IF** the unit to be purged is in MODE 5 or 6 AND the opposite unit is in MODE 1, 2, 3, or 4, **THEN**

VERIFY an operator is available and briefed on his duties related to shutdown containment purge system in accordance with 0-SO-30-3 in the event of an ABI. (This action can be performed with minimum Tech Spec shift crew without impeding mitigation of the event.) **[C.1]** ________

Unit____________ Date____________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

[15] **OPEN** the exhaust fan discharge damper **[FCV-30-213]** with **[HS-30-213]**. ________

[16] **START** the "A" Train purge/exhaust fan with **[HS-30-1A]**. ________

[17] **VERIFY** dampers **OPEN**.

"A" Train	Initials
FCO-30-1A	________
FCO-30-1B	________
FCO-30-295	________
FCO-30-294	________

[18] **OPEN** exhaust fan suction damper **[FCV-30-61]** with **[HS-30-61]**. ________

[19] **OPEN** lower containment valves **[FCV-30-14]** (supply) and **[FCV-30-56]** (exhaust) with **[HS-30-14]**. ________

[20] **OPEN** lower containment valves **[FCV-30-15]** (supply) and **[FCV-30-57]** (exhaust) with **[HS-30-15]**. ________

Unit___________ Date___________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

CAUTION **During outage conditions a flow imbalance can be created when a lower/upper compartment purge is inservice and changes are made to the upper and/or lower airlock doors (i.e. breaching or closing doors after purge is started). This situation may cause the Ice Condenser lower inlet doors to go open.**

NOTE In modes 5 or 6 only, FCV-30-16 & 17, and FCV-30-37 & 40 may be opened. However, opening them could cause an increase in containment pressure.

[21] **IF** in modes 5 or 6 **AND** additional purge flow is desired, **THEN**

OPEN the following:

A. [FCV-30-16] with [HS-30-16] ___________

B. [FCV-30-17] with [HS-30-17] ___________

C. [FCV-30-37] with [HS-30-37] ___________

D. [FCV-30-40] with [HS-30-40] ___________

[22] **OPEN** supply fan discharge damper [FCV-30-2] with [HS-30-2]. ___________

CAUTION **During fuel handling activities inside containment, purge flow shall not exceed 16,000 cfm. Containment purge flow may be monitored using FI-90-400 or using computer point Y2210A. (Reference: FSAR 15.5.6)**

[23] **IF** fuel handling is in progress or scheduled during the period this procedure section will be in effect, **THEN**

MONITOR [FI-90-400] OR [Y2210A] to ensure flow does not exceed 16,000 CFM. ___________

Unit____________ Date____________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

NOTE Lower containment purge is now in service the remaining portions of this procedure will place the Upper containment purge in service.

[24] **OPEN** one set of containment dampers with the associated handswitch. (**N/A** those not opened)

	Initials			Initials
A. Handswitch **HS-30-7** (Supply) FCV-30-7	________		A. Handswitch **HS-30-9** (Supply) FCV-30-9	________
(Exhaust) FCV-30-51	________		(Exhaust) FCV-30-53	________
AND		**OR**	**AND**	
B. Handswitch **HS-30-8** (Supply) FCV-30-8	________		B. Handswitch **HS-30-10** (Supply) FCV-30-10	________
(Exhaust) FCV-30-50	________		(Exhaust) FCV-30-52	________

Unit____________ Date____________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

CAUTION **During outage conditions a flow imbalance can be created when a lower/upper compartment purge is inservice and changes are made to the upper and/or lower airlock doors (i.e. breaching or closing doors after purge is started). This situation may cause the Ice Condenser lower inlet doors to go open.**

NOTE In modes 5 or 6 only, both sets of containment dampers may be opened. However, opening them could cause an increase in containment pressure.

[25] **IF** in modes 5 or 6, and additional purge flow is desired, **THEN OPEN**

	Initials			Initials
A. Handswitch **HS-30-7** (Supply) FCV-30-7	_________		A. Handswitch **HS-30-9** (Supply) FCV-30-9	_________
(Exhaust) FCV-30-51	_________		(Exhaust) FCV-30-53	_________
AND		**AND**	**AND**	
B. Handswitch **HS-30-8** (Supply) FCV-30-8	_________		B. Handswitch **HS-30-10** (Supply) FCV-30-10	_________
(Exhaust) FCV-30-50	_________		(Exhaust) FCV-30-52	_________

[26] **ENSURE** Lower Ice Condenser door are **CLOSED** by either of the following:

[a] Absence of Ice Condenser Lower Inlet Door Open alarm on M-6.

OR

[b] Monitoring the ice condenser door status lights on M-10 _________

Unit____________ Date____________

8.6 Simultaneous "A" Train" Purging of Lower Containment and Upper Containment (MODES 5 & 6 ONLY) (Continued)

NOTE Due to changes in the specific volume of air during purging, pressure problems may arise when equipment hatch is closed and air lock doors are breached. Additional exhaust capacity will aid in reducing Rx. Bldg. Access room pressurization.

[27] **IF** Unit is in Mode 5 or 6 and Rx. Bldg. Access room(s) are being pressurized by purge, **THEN**

PERFORM the following:

[a] **ALIGN** purge dampers as follows:

Damper	Position	Initials
FCV-30-16	CLOSED	________
FCV-30-17	CLOSED	________
FCV-30-37	OPEN	________
FCV-30-40	OPEN	________

[b] **WHEN** pressurization of Rx. Bldg. Access room(s) no longer exists (i.e. containment equipment hatch open or air locks no longer breached), **THEN**

PERFORM the following:

[i] **IF** additional purge flow was desired in step **[21]**, **THEN**

REALIGN purge dampers as follows:

Damper	Position	Initials
FCV-30-16	OPEN	________
FCV-30-17	OPEN	________

[ii] **IF** step **[21]** was N/A, **THEN**

ENSURE the following are CLOSED:

Damper	Position	Initials
FCV-30-16	CLOSED	________
FCV-30-17	CLOSED	________
FCV-30-37	CLOSED	________
FCV-30-40	CLOSED	________

END OF TEXT

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Containment Ventilation isolation system inoperable, close each of the Ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that Containment Ventilation isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

11. 025 2.2.13 001

Unit One is presently in mode 4 and preparing for a refueling outage. In order to stay ahead of the schedule you are directed to tag out the glycol expansion tank level switches.

Which one of the following devices will you use?

✓A. Plastic tag with stainless wire.

B. Paper/cloth tag with waxed twine.

C. Plastic tag with waxed twine.

D. Paper/cloth tag with stainless wire.

A. Correct, the glycol tank is inside containment. Mode 4 is greater than 200°F therefore the procedure requires a plastic tag with stainless wire if the RCS temperature is greater than 200°F.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

The student must understand the glycol tank is inside containment and that mode 4 is greater than 200°F which requires a unique card and hanging device (plastic with stainless wire).

K/A[CFR]: 025 2.2.13 [3.6/3.8] [41.10]

Reference: SPP-10.2, Tech Spec definations Table 1.1

LP/Objective: OPL271C528 B.9

History: New question.

Level: Memory

Comments: FHW 12/02 025 2.2.13

glycol tank related to 025 ice condenser?

3.4 Clearance and Tag Types

A. TVAN clearance tags shall be used for SM controlled clearances.

B. At an RCS/Moderator temperature of less than or equal to 200°F, tags (paper/cloth or plastic) that are placed inside containment may be attached to devices with either waxed twine or stainless steel wire. With an RCS/Moderator temperature in excess of 200°F, tags (plastic) that are placed inside containment shall be attached with stainless steel wire. Before exceeding an RCS/Moderator temperature of 200°F, an audit of all clearances in effect will be conducted for the purpose of identifying all tags in containment. Those tags shall be visually inspected and any tags that are paper/cloth will be replaced with plastic and any twine used to attach the tags will be replaced with stainless steel wire.

C. Tagouts may contain multiple Clearance Sections.

D. Each Clearance Section of a Tagout provides a safe clearance boundary for work on one or more discreet plant components, e.g., a pump.

E. A Clearance Section may list multiple tag types, e.g., Danger Tags, Caution Tags and Operating Permit Tags, as necessary to provide a safe clearance boundary.

F. A Clearance Section may list devices as required to support hanging the Section or to ensure restoration to normal configuration upon Section removal. Such devices do not require placement of an actual clearance tag.

G. TVAN Danger tags (TVA Form Series 19631A) must be installed on all control points that isolate equipment from sources of energy and shall bear a device identifier similar to that on the device.

H. For Caution tags, the following apply:

1. Caution tags shall be attached to equipment, switches or controls where hazardous or abnormal operating conditions exist.

2. TVAN Caution tags (TVA Form Series 19629) may be issued by the SM to indicate abnormal operating conditions.

3. Caution tags should also be used as necessary to document the need for post maintenance/modification testing (PMT) after a clearance boundary has been released.

4. A position held is not specified for Caution tags listed on a Section. The operator's signature for tag placement documents placement of the tag only. If the device must be placed in a specific position to support the placement of the clearance, specify a position in the Tag Notes for the tag.

I. For Operating Permit tags the following apply:

1. Components may be held by only one Operating Permit tag at a time.

2. Only one person at a time may sign-on as a holder of Operating Permit tags.

3. Devices held by Operating Permit tags may not be held by Danger tags at the same time.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE	
1. POWER OPERATION	≥0.99	>5%	≥350°F	R205
2. STARTUP	≥0.99	≤5%	≥350°F	
3. HOT STANDBY	<0.99	0	≥350°F	R205
4. HOT SHUTDOWN	<0.99	0	350°F > T_{avg} >200°F	
5. COLD SHUTDOWN	<0.99	0	≤200°F	
6. REFUELING**	≤0.95	0	≤140°F	R205

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

14. 056 A2.04 001

Unit One is operating at 65% power with all systems aligned normal. The crew had just placed the second Main Feedwater Pump 1 B-B in service. A ~~complete loss of~~ low pressure condition of 150 psig ~~pressure~~ developed on the common injection water supply line to the MFPs. for 10 sec and then restored.

Which one of the following is the correct result and proper action for this condition?

A. The turbine will run back and the crew should use AOP-S.01, "Loss of Normal Feedwater."

B. The unit will trip and the crew should use E-0 "Reactor Trip or Safety Injection."

C. Only the 1 B-B Main Feedwater Pump will trip and the crew should use the Annunciator Response Procedure for MFW pump "Tripped."

MFW pumps will not trip

D. There are no Main Feedwater Pump trips and the crew should use 0-GO-5, "Normal Power Operation." ~~and reduce power~~ — ID

A. Incorrect, per TI-28 attachment 9.
B. Correct per 1-AR-M3-B (A-2). the trip pressure switches are on the common supply line which will cause a trip signal to both MFPs.
C. Incorrect per 1-AR-M3-B (A-2), both MFPs will trip.
D. Incorrect per 1-AR-M3-B (A-2), both MFPs will trip.

K/A[CFR]: 056 A2.04 [2.6/2.8] [41.5 43.5]

Reference: 1,2-47W803-1
1-45N657-18
1-45N646-1
1-AR-M3-B (A-2)
TI-28 attachment 9

LP/Objective: OPL271COND.FW B.11

History: New question.

Level: Comprehension

Comments: FHW 12/02 056 A2.04

RUN ON simulator?

Source	**Setpoint**
SER 1404 R/TT-1, energized MFPT-1A tripped; SER 1406 R/TT-1, energized MFPT-1B tripped.	N/A

TRIPPED

Probable Causes

1. Thrust bearing failure.
2. Low turbine bearing oil pressure.
3. Low feedwater bearing oil pressure.
4. Main feedwater pump (MFP) isolation valve not fully open.
5. MFWP condenser vacuum low.
6. Low injection water pressure.
7. Turbine overspeed.

Corrective Actions

[1] IF unit > 80% load, THEN
VERIFY all auxiliary feedwater pumps start.

[2] IF unit > 80% load and one MFP trips, THEN
GO TO AOP-S.01, *Loss of Normal Feedwater.*

[3] IF unit < 80% load and one MFP trips, THEN
- [a] REDUCE turbine load until feedwater flow is greater than steam flow.
- [b] IF operating MFP will NOT respond in AUTO, THEN PLACE MFP master controller to MANUAL and adjust feedwater flow to greater than steam flow.
- [c] ENSURE affected MFPT condenser isolation valves CLOSED.
- [d] DISPATCH personnel to investigate trip of MFP.
- [e] ENSURE unit is returning to stable conditions.
- [f] INITIATE repairs on the affected MFP.
- [g] RETURN MFP controller to AUTO if step [b] was performed.
- [h] EVALUATE removal of secondary plant equipment based on condensate pressure.
- [i] RESET steam dumps using appropriate plant instructions, if C-7 (AR-4A, window E-5) is illuminated.
- [j] GO TO 0-GO-5, Appendix B for restoration of turbine controls.

[4] IF both MFPs trip and reactor power > 50%, THEN
VERIFY reactor tripped, AND
GO TO E-0, *Reactor Trip or Safety Injection.*

[5] IF both MFPs trip, THEN
VERIFY main turbine tripped.

References 45B655-03B-0, 45N646-1

MAIN FEEDWATER PUMP TURBINE (MFPT) TRIPS

Signal	Setpoint	Permissive
1. Manual	M-3 or Local	
2. Low Turbine Bearing Oil Pressure	< 10 psig	
3. Low Pump Bearing Oil Pressure	< 10 psig	
4. Thrust Bearing Wear Excessive	Variable (changes each time we come up in power)	Check with Instrument Dept.
5. MFP Suction Isolation Valve NOT Full Open		
6. MFP Turbine Condenser Vacuum Low	> 12.2 psia	
7. Low Injection Water Pressure	< 220 psig	20 second time delay
8. Mechanical Overspeed	6600 rpm.	
9. Feedwater Isolation Signal		Any 1 of 3 signals

TURBINE TRIPS

Signal	Setpoint	Logic	Permissive
1. Reactor Trip	From P-4 Contacts	1/2 Trains	
2. Safety Injection	SIS (K621 input)	1/2 Trains	
3. S/G Hi Hi Level (P14)	> 81%	2/3 LTs on 1/4 SGs	
4. Manual (M-2 or Local)			
5. Auto Stop Oil Pressure	< 45 psig	2/3	
6. Electrical Overspeed	110% of Rated Speed (1980 RPM)	1/1	
7. Mechanical Overspeed	110% of Rated Speed + 10 rpm (1950 - 1975 RPM)	1/1	
8. Low Vacuum	3.9 - 5.4 psia	1/1	
9. Thrust Bearing Wear	> 60 psig	1/2	
10. Low Bearing Oil Pressure	< 7 psig	1/1	
11. Stator Cooling System - Low ΔP across Stator - Stator Outlet Temp.	 12 psid less than normal ≥ 90 °C	 2/3 2/3	45 sec. time delay Gen. amps >15% Gen. amps >15%
12. High Vibration	> 14 mils (Alarm @ > 7 mils)	1/11	Not Cutout
13. MFPT A & B Tripped		2/2	
14. Main Generator Input Trips - Reverse Power (32X)* - Gen. Backup (86GBX) - Neg. Phase Sequence (46X) - Overcurrent (51GX) - System Ground (51/59X) - Differential (87X)			* 30 sec. time delay
15. Main Transformer Input Trips - Differential (87TX) - Low Voltage Bushing Oil Flow Low		 1/3 1/3 (on 1/3 transformers)	Unit 1 only.
16. USS Transformer A & B Input Trips - Differential (A87SX) - Differential (B87SX)			
17. Loss of ± 48 V DC		2/2	
18. Loss of ± 15 V DC		2/2	

15. 056 K1.03 001

Unit One is operating at 100% power and all systems are aligned normal. The crew notices the "PS-2-129 Low NPSH at MFP's" alarm just annunciated.

Which one of the following could bring in this alarm?

A. Condensate low pressure heater strings "A" and "B" isolated on hi-hi #7A and #7B heater levels.

B. High pressure feedwater heater #1A discharge valve was inadvertently closed.

✓C. Trip of a condensate booster pump.

D. Condensate intermediate heaters #3B and #3C are above program level.

A. Incorrect, there is no string isolation logic associated with the #7 heaters.
B. Incorrect, the isolated #1 heater discharge valve is downstream of the MFP.
C. Correct, per reference.
D. Incorrect, the #3 heaters do not have a program level.

K/A[CFR]: 056 K1.03 [2.6/2.6] [41.2-9]

Reference: 1-AR-M3-A (E-1)
1,2-47W611-2-1
1,2-47W611-2-2
1,2-47W803-1

LP/Objective: OPN218SS.006A B.3

History: New question.

Level: Memory

Comments: FHW 12/02 056 K1.03

Similar situation to Question 14

Source	Setpoint	
SER 214 PS-2-129A	100 psid decreasing (differential between MFP inlet and #2 heater shell)	PS-2-129 LOW NPSH AT MFP'S

Probable Causes

1. Increasing load.
2. Condensate booster pump trip.

Corrective Actions

[1] ENSURE MFP inlet pressure greater than 320 psig by reducing turbine load [M-3, PI-2-129].

[2] ENSURE the following pumps are operating as required by 0-GO-5 for the present unit load.

a. Hotwell pumps
b. Condensate DI booster pumps
c. Condensate booster pumps
d. #7 HDT pumps
e. #3 HDT pump

[3] ADJUST condensate and feedwater pressure as required to clear alarm.

[4] IF MFP trips, THEN
GO TO AOP-S.01, *Loss of Normal Feedwater.*

References

47B601-2-21
45B655-03A-2
47W611-2-2

17. 059 AA2.05 001

Given the following plant conditions and information:

CONDITIONS

- Unit 1 is at 210°F and is being cooled down after discovery of a fuel clad failure.
- RCDT pump A is being used to lower level in the RCDT.
- Panel 1-XA-55-30 window "1-RA-277A RCDT Hi Rad" is in alarm.

INFORMATION

- 1-FCV-77-9 (inside containment isolation valve).
- 1-FCV-77-10 (outside containment isolation valve).

The SRO directs the Operator to pump the contents of the RCDT to the TDCT. Which of the following actions should the Operator take to pump down the RCDT?

A. Place 1-HS-77-9B and 1-HS-77-10B to "Normal" then open 1-FCV-77-9A and 1-FCV-77-10A.

✓B. Place 1-HS-77-9B and 1-HS-77-10B to "Block" then open 1-FCV-77-9A and 1-FCV-77-10A.

C. Place 1-HS -77-9A and 1-HS-77-10A to "Open."

D. Cycle1-HS-77-9A and1-HS-77-10A back to "A-Auto."

A. Incorrect per reference.
B. Correct per reference, block position will override a high rad signal.
C. Incorrect per reference, will not override a high rad signal.
D. Incorrect per reference.

K/A[CFR]:	059 AA2.05 [3.6/3.9] [43.5]
References:	47W611-77-1 1-AR-M30-A (B-3)
LP/Objectives:	OPL271RADMON B.8
History:	New question
Level:	Memory
Comments:	FHW 12/02 059 AA2.05

How is this a release? otherwise K/A is ok

Source	Setpoint	
SER 1739 1-RE-90-277	316 mR/Hr	1-RA-277A RCDT HI RAD

Probable Causes

1. Fuel cladding failure.
2. High activity in reactor coolant.

Corrective Actions

[1] VERIFY [1-FCV-77-9] RCDT pump to TDCT isolation valve closes.

[2] CHECK [1-RM-90-277] for high radiation on panel 1-M-30.

[3] IF need to pump RCDT to FDCT with this high radiation condition present, THEN

REQUEST permission from SRO to place the following hand switches in block:

a. [1-HS-77-9B]

b. [1-HS-77-10B]

[4] WHEN ready to pump RCDT, THEN

OPEN [1-FCV-77-10] and .[1-FCV-77-9].

[5] REQUEST radiochemical laboratory sample to verify activity.

[6] IF high activity verified, THEN

GO TO AOP-R.06, *High RCS Activity.*

References

47B601-55-75, 47W610-90-4

26. 078 K3.03 001✓

Plant Conditions:
- Unit One is operating at 100% power.
- Unit Two is operating at 100% power.
- Auxiliary Control Air Compressor A-A power supply breaker on 480V C&A Vent Bd 2A1-A is tagged.

A pipe break in the control air header has allowed the control air header pressure to drop to 65 psig.

The Auxiliary Control Air Compressor B-B breaker on 480V C&A Vent BD 2B1-B has tripped.

Which one of the following is correct for these conditions.

A. Unit Two equipment using "B" train instrument air only is ~~e~~affected.

B. Unit Two equipment only using both trains of instrument air are ~~e~~affected.

C. Unit One equipment only using both trains of instrument air are ~~e~~affected.

✓D. Unit One and Unit Two equipment using both trains of instrument air are ~~e~~affected.

A. Incorrect, control air compressors to auxiliarly air compressors isolate when control air pressure reaches 69 psig and neither aux air compressors are available therefore both trains of air are effected.
B. Incorrect, control air compressors to auxiliarly air compressors isolate when control air pressure reaches 69 psig and neither aux air compressors are available therefore both trains of air are effected which also supply Unit one.
C. Incorrect, control air compressors to auxiliarly air compressors isolate when control air pressure reaches 69 psig and neither aux air compressors are available therefore both trains of air are effected.
D. Correct, the auxiliary control air compressors supply both units.

K/A[CFR]:	078 K3.03 [3.0/3.4] [41.7]
Reference:	AOP-M.02 1,2-47W848-1 1,2-47W848-12
LP/Objective:	OPL271CSA B.11
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 078 K3.03

4 correct answers

3.1 Symptoms (cont'd)

A. Deviations or unexpected indications on any of the following may indicate a loss of Control Air System pressure:

- Aux Control Air Header pressure dropping:
  - 0-PI-32-104A, Aux Control Air Hdr A Press [M-15].
  - 0-PI-32-105A, Aux Control Air Hdr B Press [M-15].
- Control and Service (C&S) air compressors tripped.
- Valves or dampers moving to failed position(s).
- S/G anomalies due to MFW Reg Valves failing closed.
- MSIVs fail closed.

B. Any of the following automatic actions may indicate a loss of Control Air System pressure:

SETPOINT (psig)	RANGE (psig)		INITIATING EVENT	EVENT
	min	max		
N/A	--	--	Normal control circuit failure	C&S air compressors auto start with backup control circuit
88	86	90	Control Air Receiver dropping	Service Air isolates from Control Air (0-PCV-33-4)
				C&S air compressors load to 50%
86	84	88	Control Air Receiver pressure dropping	C&S air compressors load to 100%
77	74.5	79.5	Aux Air Receiver pressure dropping	Aux Air Compressors start
69	66.5	71.5	Control Air header pressure dropping	Aux Air isolates from Control Air (0-FCV-32-82 and 85)
50	--	--	Aux Air Receiver pressure dropping	Aux Air to containment valves fail closed. (1-FCV-32-80, -102, -110, 2-FCV-32-81, -103, -111)

3.2 Entry Conditions

None

27. 079 2.1.1 001

The "Control and Service Air Compessor C or D Trouble/Shutdown" alarm just annunciated.

Which of the following is correct according to conduct of operations requirements.

A. Use the PA to dispatch operations personnel to the air compressors. — possibly correct

B. Use two way communications since there is no urgency. — not relevant

C. Acknowledge the alarm and assume it was a pre-planned activity. — would be ok if stem addressed maintenance activity

✓D. Acknowledge the alarm and perform actions in the annunciator response.

A. Incorrect per reference.
B. Inorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A[CFR]:	079 2.1.1 [3.7/3.8] [41.10]
Reference:	OPDP-1 R1
LP/Objective:	OPL271C209 B.6 and 8
History:	New question.
Level:	Memory
Comments:	FHW 12/02 079 2.1.1

DISCUSS

3.10 Communications

3.10.1 Public Address System

Use of the plant public address system (page) should be administratively controlled to ensure it retains its effectiveness in contacting plant personnel. When using the PA system:

A. Speak slowly and deliberately in a normal tone of voice, using normally accepted equipment nomenclature and phonetic alphabet designations where needed to preclude potential confusion.

B. When announcements of abnormal or emergency conditions are made , they shall be made at least twice.

C. When announcing drill exercises, precede and follow all communications with, "This is a drill."

The following activities require a PA announcement:

A. Planned starting or tripping of large or loud equipment.
B. Change of plant status, i.e., unit scram/trip.
C. To summons operators or other plant personnel as needed.
D. Ventilation alignment changes which may temporarily affect access and egress to plant areas.

3.10.2 Radios

Radio usage shall not be allowed in areas where electronic interference with plant equipment may result. Areas where radio use is prohibited shall be posted. When using radios:

A. Identify yourself and your watch station.
B. Utilize three-way communication technique, normally accepted equipment nomenclature, and phonetic alphabet designations where appropriate.
C. Use professional language.
D. If signal breakup is experienced, relocate or use alternate method of communication.
E. When announcing drill exercises, precede and follow all communications with, "This is a drill."

3.10.3 Telephone

Telephones are the preferred method of remote communication. When using telephones:

A. Identify yourself and your watch station.
B. Utilize three-way communication technique, where appropriate.
C. Use professional language.
D. When announcing drill exercises, precede and follow all communications with, "This is a drill."

28. 086 K3.01 001

The fire protection system failed to actuate allowing a spreader room fire to burn control wiring. This initiated a control room abandonment requiring the Operators to complete AOP-C.04 "Control Room Inaccessibility" checklists.

Which one of the following is correct for this condition? (Assume checklists completed.)

A. The 1 A-A MDAFW pump can be started from the Unit 1 Auxiliary Control Room. ?

B. The 2 B-B RHR pump can be started from the Unit 2 Auxiliary Control Room.

C. The 1 B-B Safety Injection pump can be started from the 480v Shutdown Bd.1B1-B.

✓D. The 2 B-B RHR pump can be started from the 6.9 kv Shutdown Bd. 2 B-B.

A. Incorrect, the transfer of control was completed when the AOP-C.04 checklists were finished
B. Incorrect, electrical devices are controlled from their switchgear.
C. Incorrect, the pump is supplied from the 6.9 kv shutdown bd.
D. Correct, the electrical devices are controlled from their switchgear.

K/A[CFR]:	086 K3.01 [2.7/3.2] [41.7]
Reference:	AOP-C.04
LP/Objective:	OPL271C423 B.5
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 086 K3.01

explain A

CHECKLIST 3

6.9-kV SHUTDOWN BOARD 1B-B

NOTE To ensure one ERCW pump will sequence on following a blackout, its auxiliary control switch must be placed in START. If an ERCW pump is running (breaker closed), it should be selected.

[7] **PLACE** auxiliary control switch for one ERCW pump in START momentarily. [Compt 8 or 9] ☐

[8] **ENSURE** transfer switches in AUX position:

EQUIPMENT NAME	COMPT	TRANSFER SWITCH	SWITCH POSITION
EMERG SUPPLY BRKR FROM D/G 1B-B BREAKER 1914	6	1-XS-57-73	☐ AUX
ESSENTIAL RAW CLG WATER PUMP L-B	8	0-XS-67-440	☐ AUX
ESSENTIAL RAW CLG WATER PUMP N-B	9	0-XS-67-452	☐ AUX
AUX FEEDWATER PUMP 1B-B	10	1-XS-3-128	☐ AUX
NORM SUPPLY BRKR FROM 6.9KV UNIT BD 1C BREAKER 1726	11	1-XS-57-68	☐ AUX
CONTAINMENT SPRAY PUMP 1B-B	13	1-XS-72-10	☐ AUX
RESID HEAT REMOVAL PUMP 1B-B	14	1-XS-74-20	☐ AUX
SAFETY INJECTION PUMP 1B-B	15	1-XS-63-15	☐ AUX
ALT SUPPLY BRKR FROM 6.9KV UNIT BD 1D BREAKER 1728	16	1-XS-57-71	☐ AUX
CENTRIFUGAL CHARGING PUMP 1B-B	18	1-XS-62-104	☐ AUX
PZR HEATERS BACKUP GROUP 1B-B	20	1-XS-68-341D	☐ AUX
PZR HEATER BACKUP GROUP 1C	21	1-XS-68-341H	☐ AUX

[9] **ENSURE** <u>[0-XS-82-122]</u> GEN 1B-B TRANSFER switch in AUX position [Shutdown Bd 1B-B Logic Panel 4] ☐

CHECKLIST 3

6.9-kV SHUTDOWN BOARD 1A-A

NOTE To ensure one ERCW pump will sequence on following a blackout, its auxiliary control switch must be placed in START. If an ERCW pump is running (breaker closed), it should be selected.

[4] **PLACE** aux control switch for one Train A ERCW pump in START momentarily. [Compt 8 or 9] ☐

[5] **ENSURE** transfer switches in AUX position:

EQUIPMENT NAME	COMPT	TRANSFER SWITCH	AUX POSITION
EMERGENCY SUPPLY BREAKER FROM DSL GENERATOR 1A-A BREAKER 1912	6	1-XS-57-46	☐ AUX
ESSENTIAL RAW CLG WATER PUMP J-A	8	0-XS-67-432	☐ AUX
ESSENTIAL RAW CLG WATER PUMP Q-A	9	0-XS-67-460	☐ AUX
AUXILIARY FEEDWATER PUMP 1A-A	10	1-XS-3-118	☐ AUX
NORMAL SUPPLY BREAKER FROM 6900 V UNIT BOARD 1B BREAKER 1718	11	1-XS-57-44	☐ AUX
CONTAINMENT SPRAY PUMP 1A-A	13	1-XS-72-27	☐ AUX
RESIDUAL HEAT REMOVAL PMP 1A-A	14	1-XS-74-10	☐ AUX
SAFETY INJECTION PUMP 1A-A	15	1-XS-63-10	☐ AUX
ALTERNATE SUPPLY BREAKER FROM 6900 V UNIT BOARD 1A BREAKER 1716	16	1-XS-57-41	☐ AUX
CENTRIFUGAL CHARGING PUMP 1A-A	18	1-XS-62-108	☐ AUX
PZR HEATERS BACKUP GROUP 1A-A	20	1-XS-68-341A	☐ AUX
PZR HEATERS GROUP 1D	21	1-XS-68-341F	☐ AUX

[6] **ENSURE** <u>[0-XS-82-121]</u> GEN 1A-A TRANSFER switch in AUX position [Shutdown Bd 1A-A Logic Relay Panel 4] ☐

CHECKLIST 4

6.9-kV SHUTDOWN BOARD 2B-B

NOTE To ensure one ERCW pump will sequence on following a blackout, its auxiliary control switch must be placed in START. If an ERCW pump is running (breaker closed), it should be selected.

[7] **PLACE** auxiliary control switch for one ERCW pump in START momentarily. [Compt 8 or 9] ☐

[8] **ENSURE** transfer switches in AUX position:

EQUIPMENT NAME	COMPT	TRANSFER SWITCH	SWITCH POSITION
EMERGENCY SUPPLY BREAKER FROM DSL GENERATOR 2B-B BREAKER 1924	6	2-XS-57-73	☐ AUX
ESSENTIAL RAW CLG WATER PUMP P-B	8	0-XS-67-456	☐ AUX
ESSENTIAL RAW CLG WATER PUMP M-B	9	0-XS-67-444	☐ AUX
AUX FEEDWATER PUMP 2B-B	10	2-XS-3-128	☐ AUX
NORMAL SUPPLY BREAKER FROM 6900 V UNIT BOARD 2C BREAKER 1826	11	2-XS-57-68	☐ AUX
CONTAINMENT SPRAY PUMP 2B-B	13	2-XS-72-10	☐ AUX
RESID HEAT REMOVAL PUMP 2B-B	14	2-XS-74-20	☐ AUX
SAFETY INJECTION PUMP 2B-B	15	2-XS-63-15	☐ AUX
ALTERNATE SUPPLY BREAKER FROM 6900 V UNIT BOARD 2D BREAKER 1828	16	2-XS-57-71	☐ AUX
CENTRIFUGAL CHARGING PUMP 2B-B	18	2-XS-62-104	☐ AUX
PRESSURIZER HEATERS BACKUP GROUP 2B-B	20	2-XS-68-341D	☐ AUX
PRESSURIZER HEATER BACKUP GROUP 2C	21	2-XS-68-341H	☐ AUX

[9] **ENSURE** [0-XS-82-124] GEN 2B-B TRANSFER switch in AUX position [Shutdown Bd 2B-B Logic Panel 4] ☐

30. 2.1.16 001

The Unit One operator notices the service air receiver is isolated and control air pressure is dropping.

Which of the following is correct for announcements of this abnormal condition?

A. Announce over the PA system one time only and phone the AUO field office.

✓B. Announce at least twice over the PA system.

C. Announce over the PA system one time. only

D. ~~Use the added phrase, "This is **NOT** a drill."~~ Announce over the radio using 2-way communication.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]:	2.1.16 [3.7/3.8] [41.10]
Reference:	OPDP-1 R1 section 3.10.1
LP/Objective:	OPL271C209 B.6
History:	New question.
Level:	Memory
Comments:	FHW 12/02 2.1.16

~~reference does not support answer~~

3.10 Communications

3.10.1 Public Address System

Use of the plant public address system (page) should be administratively controlled to ensure it retains its effectiveness in contacting plant personnel. When using the PA system:

A. Speak slowly and deliberately in a normal tone of voice, using normally accepted equipment nomenclature and phonetic alphabet designations where needed to preclude potential confusion.

B. When announcements of abnormal or emergency conditions are made , they shall be made at least twice.

C. When announcing drill exercises, precede and follow all communications with, "This is a drill."

The following activities require a PA announcement:

A. Planned starting or tripping of large or loud equipment.
B. Change of plant status, i.e., unit scram/trip.
C. To summons operators or other plant personnel as needed.
D. Ventilation alignment changes which may temporarily affect access and egress to plant areas.

3.10.2 Radios

Radio usage shall not be allowed in areas where electronic interference with plant equipment may result. Areas where radio use is prohibited shall be posted. When using radios:

A. Identify yourself and your watch station.
B. Utilize three-way communication technique, normally accepted equipment nomenclature, and phonetic alphabet designations where appropriate.
C. Use professional language.
D. If signal breakup is experienced, relocate or use alternate method of communication.
E. When announcing drill exercises, precede and follow all communications with, "This is a drill."

3.10.3 Telephone

Telephones are the preferred method of remote communication. When using telephones:

A. Identify yourself and your watch station.
B. Utilize three-way communication technique, where appropriate.
C. Use professional language.
D. When announcing drill exercises, precede and follow all communications with, "This is a drill."

33. 2.3.10 001

A Control Room Isolation was automatically initiated during normal power operations.

Plant Conditions:

- Control Bldg Emergency Air Cleanup Fan A-A is running.
- Control Bldg Emergency Air Cleanup Fan B-B is running.
- Control Bldg Emergency Pressurization Fan A-A is running.
- Control Bldg Emergency Pressurization Fan B-B is **NOT** running.
- Spreading Room Supply Fan is **NOT** running.
- Spreading Room Exhaust Fans A and B are **NOT** running.
- Locker Room Exhaust Fan is **NOT** running.
- 0-FCV-311-105A (MCR fresh air inlet) is open.
- 0-FCV-311-106A (MCR fresh air inlet) is open.

Which one of the following is correct for this condition?

✓A. Close 0-FCV-311-105A and 106A to limit potential personnel exposure ~~to 5 Rem whole body.~~

B. 0-FCV-311-105A and 106A should ~~be open~~ remain open to ensure control room will remain habitable.

C. Spreader Room Supply, Spreader Room Exhaust Fans, and the Locker Room exhaust fans should be running to ensure control room will remain habitable.

D. Stop Control Bldg Emergency Pressurization Fan A-A since Control Bldg Emergency Air Cleanup Fan A-A is running maintaining Control Room environment.

A. Correct per references.
B. Incorrect, MCR fresh air inlets should be closed.
C. Incorrect, these fans should be stopped.
D. Incorrect, at least one emergency air pressurizing fan should be running.

K/A[CFR]:	2.3.10 [2.9/3.3] [43.4]
Reference:	0-SO-30-2 R9 Tech Specs bases B3/4.7.7.
LP/Objective:	OPL271CBVENT B.11
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 2.3.10

Not discriminating since a is very obvious. Better to test knowledge of pressurizing fan

3/4.7.5 ULTIMATE HEAT SINK (UHS)

The limitations on UHS water level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits. R83

The limitations on the maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants", March 1974. R12

The limitations on minimum water level are based on providing sufficient flow to the ERCW serviced heat loads after a postulated event assuming a time-dependent drawdown of reservoir level. Flow to the major transient heat loads (CCS and CS heat exchangers) is balanced assuming a reservoir level of elevation 670. The time-independent heat loads (ESF room coolers, etc.) are balanced assuming a reservoir level of elevation 639. R83 BR-10

3/4.7.6 FLOOD PROTECTION

This specification is deleted. R251

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

Date__________

8.1 Emergency Mode Control Room Isolation (Continued)

[7] **ENSURE** either Control Building Emergency Air Cleanup fan **RUNNING** and associated fan inlet **OPEN**:

CONTROL BLDG EMERGENCY AIR CLEANUP FAN	RUNNING √	FAN INLET	OPEN √
A	☐	0-FCO-311-9	☐
B	☐	0-FCO-311-11	☐

[8] **ENSURE** at least one Emergency Air Pressurizing Fan **RUNNING** and associated fan inlet **OPEN**:

CONTROL BLDG EMERGENCY PRESSURIZING FAN	RUNNING √	FAN INLET	OPEN √
A	☐	0-FCO-311-108	☐
B	☐	0-FCO-311-109	☐

[9] **ENSURE** MCR and Spreading Room Fresh Air Fans **STOPPED**:

[a] Spreading Room Supply Fan. ☐

[b] Spreading Room Exhaust Fan A. ☐

[c] Spreading Room Exhaust Fan B. ☐

Date__________

8.1 Emergency Mode Control Room Isolation (Continued)

[10] **ENSURE** MCR and Spreading Room Fresh Air Dampers **CLOSED**:

DAMPER	DESCRIPTION	CLOSED √
0-FCV-311-105A	MCR fresh air	☐
0-FCV-311-106A	MCR fresh air	☐
0-FCV-311-105B	Spreading room fresh air	☐
0-FCV-311-106B	Spreading room fresh air	☐
0-FCO-311-79	Spreading Room Exhaust Fan A outlet	☐
0-FCO-311-80	Spreading Room Exhaust Fan B outlet	☐
0-FCO-311-17	Spreading room supply discharge	☐
0-FCO-311-102	Spreading room supply discharge	☐

[11] **ENSURE** Locker Room Exhaust Fan **STOPPED**. ☐

[12] **ENSURE** Locker Room Exhaust Dampers **CLOSED:**

[a] **[0-FCO-311-103]**,Toilet and Locker Room Exhaust Fan Discharge. ☐

[b] **[0-FCO-311-104]**,Toilet and Locker Room Exhaust Fan Discharge. ☐

NOTE Battery Room Exhaust Fans are started and stopped, via their respective breakers on the 480V C&A Vent Boards.

[13] **IF** one Electrical Board Room AHU in service, **THEN**

ENSURE one of the following Battery Room Exhaust Fans **RUNNING**:

[a] Battery Room Exhaust Fan A. [C&A Vent Board 1A1-A / 12A] ☐

[b] Battery Room Exhaust Fan B. [C&A Vent Board 1B1-B / 11E] ☐

[c] Battery Room Exhaust Fan C. [C&A Vent Board 2B1-B / 11E] ☐

38. AOP-C.04 001

Given the following plant conditions:

- A fire occurs in the cable spreading room while both units are at 100% power.
- The operating crew places HS-13-204 and 205, Train A and B MOV Shunt Trip, to the TRIP position.

Which ONE (~~1~~) of the following explains why this action is performed?

✓A. Remove power from certain critical valves to prevent inadvertent operation.

Transfer control of vital equipment to local control stations

B. Initiate the fire suppression equipment (CO_2) to the cable spreading room. —ID

C. Remove control power from certain critical valves to prevent operation either from the main control room or from the switchgear.

D. Trip the spreading room ventilation equipment to compartmentalize the fire and to keep it from spreading.

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per references.
D. Incorrect per reference.

2-47W611-74-1 shows that the shunt trip breaker will "stop" valve motor when in "normal" but will not when in "auxiliary," therefore distractor "C" is incorrect.

K/A[CFR]: 067 AA2.17 [3.5/4.3] [43.5]

Reference: AOP-C.04, page 5; 1-47W611-74-1

Objective: OPL271C423, B.1

Level: Memory

History: Procedure bank

Comments: FHW 12/02 067 AA2.17✓/ RM

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Control Room Abandonment (cont'd)

NOTE The following step trips shunt trip breakers for thermal barrier isolation valves on both units and various ERCW and CCS valves.

6. **ENSURE** the following handswitches placed in TRIP: [1-M-15]
   - 0-HS-13-204
   - 0-HS-13-205

7. **ANNOUNCE** "Unit _____ Reactor trip, abandoning the Main Control Room" **USING** PA System.

NOTE Radio use is allowed in ACR during an emergency.

8. **ENSURE** the following items are taken to Auxiliary Control Room when Main Control Room is evacuated:
   - flow prints
   - radios

9. **EVACUATE** Main Control Room on affected unit(s).

41. AOP-M.01-B.3 002
With a rupture of the ERCW return header in the Auxiliary Building, during normal full power operation, how long may the CCP and SI pumps operate without experiencing bearing failure after the loss of ERCW cooling?

A. 1 hour

B. 45 minutes

C. ~~Indefinitely~~ 30 minutes

✓D. 10 minutes

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A[CFR]: 2.4.20 [3.3/4.0] [41.10]

Reference: AOP-M.01 section 2.3.

LP/Objective: OPL271AOPM01, B.3

History: Procedure bank, old Bank Number B-1090

Level: Memory

Comment: FHW 12/02 2.4.20

EOP warnings cautions & Notes
This is AOP Related.
Have to use argument of AOP part of EOP network.

1. Discuss leak path

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.3 ERCW Supply Header 1B Failure to Auxiliary Building

CAUTION: During operation, CCP and SI Pumps may experience bearing failure 10 minutes after loss of ERCW cooling.

1. **DISPATCH** personnel to locate failure.

2. **DISPATCH** operators with radios to PERFORM:
Appendix F,
Rx MOV Board Appendix R ERCW Valves
[Aux Bldg, 749' elev, Rx MOV Boards].

3. **ENSURE** 1A CCP RUNNING.

4. **STOP** and **LOCK OUT** the following:
   - 1B CCP
   - 1B SI Pump

5. **START** additional Lower Compartment Cooling Fans and CRDM Fans as required to maintain containment temperature.

57. D/G-B.6 012

Diesel Generator 1A-A was emergency started from the backup control room during an evacuation of the main control room. Which ONE ~~(1)~~ of the following conditions would result in a trip of the Diesel Generator 1A-A?

✓A. Diesel Generator 1A-A Low lube oil pressure actuates ~~decreases and remains at 2 psig.~~

B. Diesel Generator 1A-A phase imbalance relay actuates.

C. Diesel Generator 1A-A reverse power relay actuates.

D. Diesel Generator 1A-A neutral overcurrent relay actuates.

A Correct while emergency start switches on O-M-26 and O-L-4 are held in emergency start position, all engine trips are defeated except for overspeed. When the switches are released to normal, the emergency-start relay is reenergized and all engine protection is restored.
B. Incorrect, the only generator protection is differential overcurrent.
C. Incorrect, the only generator protection is differential overcurrent.
D. Incorrect, the only generator protection is differential overcurrent.

K/A[CFR]:	064 K4.02 (3.9-4.2) {41.7} 068 AK2.07 [3.3/3.4] {41.7}
Reference:	0-SO-82-1 0-45N767-2
LP/Objective:	OPL271C065, b.6
History:	System bank old Bank Number PL-1426
Level:	Memory
Comment:	FHW 12/02 068 AK2.07✓ OK

what is actual L.O. pressure setpoint?

APPENDIX A
Page 1 of 1

STANDBY MODE PARAMETERS

PARAMETER	INSTRUMENT	LIMITS MIN	LIMITS MAX	INITIALS
ENGINE 1A1				
Lube Oil Temperature to Engine	0-TI-82-5010/1	85°F	125°F	
Lube Oil Crankcase Level	Dipstick	Low Mark	High Mark	
Lube Oil Circulating Pump	--	Running		
Circulating Lube Oil Pressure	0-PI-82-5016/1	10 psig	--	
Jacket Water Immersion Heater	Disconnect Switch	Heater ON (1)		
Jacket Water Temperature to Engine	0-TI-82-5006/1	100°F	125°F	
Cooling Water Expansion Tank Level	0-LI-82-5004/1	Min Stop Mark	Max Stop Mark	
Woodward Governor Oil Level	Sightglass	Low Mark	--	
Day Tank #1 Level	0-LI-18-61/1	250 gals (2)	500 gals	
ENGINE 1A2				
Lube Oil Temperature to Engine	0-TI-82-5008/1	85°F	125°F	
Lube Oil Crankcase Level	Dipstick	Low Mark	High Mark	
Lube Oil Circulating Pump	--	Running		
Circulating Lube Oil Pressure	0-PI-82-5015/1	10 psig	--	
Jacket Water Immersion Heater	Disconnect Switch	Heater ON (1)		
Jacket Water Temperature to Engine	0-TI-82-5003/1	100°F	125°F	
Cooling Water Expansion Tank Level	0-LI-82-5001/1	Min Stop Mark	Max Stop Mark	
Woodward Governor Oil Level	Sightglass	Low Mark	--	
Day Tank #2 Level	0-LI-18-76/1	250 gals (2)	500 gals	
Seven Day Tank Level	0-LI-18-38	4.7 ft. (2)	5.1 ft.	

(1) Not required for operability if lube oil temperature is greater than 85°F.

(2) Tech Spec limit. Notify Unit SRO if level is approaching or is below the limit.

60. E01 2.2.25 001 ~~[illegible]~~ Keep in past tense

Plant Conditions:

- CCP "A" is tagged for motor replacement.
- A small RCS leak occurs with RCS pressure dropping to 1950 psig.

~~[illegible]~~ ~~because there was~~

The procedure reader ~~gets confused~~ got concerned while reading E-0 step four ~~and is concerned about having~~ only one train of ECCS ~~during a LOCA.~~ available.

The procedure reader used his own judgement and transitioned to ES-0.0.

Which one of the following is correct concerning his actions and concerns?

✓A. His transition was incorrect ~~and~~ Because TS bases for ECCS is based on a LOCA. with 1 train availa[ble]

B. His transition was correct ~~and~~ because TS bases for ECCS is based on a LOCA. "

C. His transition was incorrect ~~and~~ because TS bases for ECCS is based on a SG tube rupture. "

D. His transition was correct ~~and~~ because TS bases for ECCS is based on a SG tube rupture. "

A. Correct, he should not go to ES-0.0 since there was no SI (RCS pressure dropped to 1950 psig), and TS bases for ECCS is a LOCA.
B. Incorrect, he should not go to ES-0.0 since there was no SI, and TS bases for ECCS is a LOCA.
C. Incorrect, he should not go to ES-0.0 since there was no SI, and TS bases for ECCS is a LOCA.
D. Incorrect, he should not go to ES-0.0 since there was no SI, and TS bases for ECCS is a LOCA.

K/A[CFR]: E01 2.2.25 2.5/3.7] [43.2]

Reference: EPM-4 section 3.11.5 and TS bases 3/4.5.2 and 3/4.5.3.

LP/Objective: OPL271C266 B.3

History: New question.

Level: Comprehension

Comments: FHW 12/02 E01 2.2.25

confused is specific determined indicating action was wrong

3.11.5 Use of ES-0.0, Rediagnosis

A. ES-0.0, *Rediagnosis*, is unique among the EOPs in that it has no specific transition into it. It is entered strictly based on operator judgment and is applicable only if SI is in progress and E-0 has already been performed.

B. ES-0.0 should be used when the operator has any concern that he may not be in the right EOP based on plant conditions. This is most likely to happen if multiple accidents occur either simultaneously or sequentially.

C. Once entered, ES-0.0 will either transition the operator to ECA-2.1, E-1, E-2, or E-3, or will return him to the procedure and step in effect, depending on diagnostics done within the procedure.

D. If ES-0.0 determines that an operator should be in a certain series of procedures (e.g., E-1 or ECA-1 series), and he is, then he simply returns to the procedure and step in effect.

E. If ES-0.0 determines that an operator should be in a certain series of procedures (e.g., E-3 or ECA-3 series), and he is NOT, then he is sent to either E-1 (if he should be in E-1 or ECA-1 series) or E-3 (if he should be in E-3 or ECA-3 series) to enter the appropriate series at the beginning and work his way through the series normally from that point on.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each cold leg injection accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core in the event that the RCS pressure falls below the specified pressure of the accumulators. For the cold leg injection accumulators, this condition occurs in the event of a large or small rupture. R144 R144

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The limits in the specification for accumulator nitrogen cover pressure are analysis limits and do not include instrument uncertainty. The cold leg accumulator volume (level) values in the limiting condition for operation, TS 3/4.5.1, are the operating limits. The analysis limits bound the operational limits with instrument uncertainty applied. The minimum boron concentration ensures that the reactor core will remain subcritical during the post-LOCA (loss of coolant accident) recirculation phase based upon the cold leg accumulators' contribution to the post-LOCA sump mixture concentration. BR-13 R159

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except boron concentration not within limits minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. Under these conditions, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required. For an accumulator inoperable due to boron concentration not within limits, the limits for operation allow 72 hours to return boron concentration to within limits. This is based on the availability of ECCS water not being affected and an insignificant effect on core subcriticality during reflood because boiling of ECCS water in the core concentrates boron in the saturated liquid. R196

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

62. E15 EK2.2 001

Plant Conditions:

- The unit is recovering from a LOCA.
- Containment pressure peaked at ~~2.5~~ 12 psid. ?
- RWST level is 3%.
- Containment sump level is 60%.

All actions due to initial conditons have been completed. Which one of the following best describes the core heat removal?

A. Core heat is removed through the break and by ECCS injection from the RWST.

✓B. Core heat is removed ~~through the break~~ and the RHR heat exchangers.

C. Core heat is removed through the break and containment spray heat exchangers.

D. Core heat is removed through the RHR and containment spray heat exchangers.

A. Incorrect, the RWST is 3% and sump level is 60% therefore sump swapover should have occurred.

B. Correct, heat removal is from the break and from the initial conditions the operators should have performed ES-1.3 therefore RHR heat exchangers are removing core heat.

C. Incorrect, the containment pressure never reached 2.81 psid which requires containment sprays.

D. Incorrect, the containment pressure never reached 2.81 psid which requires containment sprays.

K/A[CFR]: E15 EK2.2 [2.7/2.9] [41.7]

Reference: ES-1.3
EPM-3-ES-1.3 step 5

LP/Objective: OPL271C388 B.2

History: New question.

Level: Comprehension

Comments: FHW 12/02 E15 EK2.2

IF pressure only peaked at 2.5 psig then break is small AND this got out of control. Core heat removal through small break? check on simulator

Better to have LBLOCA with pressure mitigation AND sprays no longer necessary by procedure.

1.0 PURPOSE

This procedure provides the necessary instructions for transferring the safety injection system and containment spray system to the recirculation mode.

2.0 SYMPTOMS AND ENTRY CONDITIONS

2.1 ENTRY CONDITIONS

E-1 Loss of Reactor or Secondary Coolant:

- RWST level less than 27%.

ECA-2.1 Uncontrolled Depressurization of All Steam Generators:

- RWST level less than 27%.

Other EOPs:

- When RWST level reaches the switchover setpoint.

3.0 OPERATOR ACTIONS

EOP Step Number: 5

ESTABLISH CCS to RHR heat exchangers:

ERG Step Number: 2

Verify CCW Flow To RHR Heat Exchangers:

Purpose:

To ensure CCS flow to cool the recirculation fluid.

ERG Basis:

This step assumes that the RHR heat exchangers are used for heat removal during the post-accident recirculation phase and that either CCS flow has been automatically provided to the heat exchangers or the operator has manually established CCS flow prior to the switchover alarm. If CCS flow had not previously been established, then it should be established at this time.

Knowledge:

If CCS cannot be established to one heat exchanger, the remaining guideline can be performed as listed provided that the uncooled recirculation fluid temperature and pressure do not exceed equipment design conditions.

EOP Basis:

Same.

Deviation:

Made high level step ESTABLISH versus the ERG's AER:VERIFY/RNO:ESTABLISH arrangement.

Justification:

Improves readabililty and is more direct when implementing plant-specific details.

Setpoint:

Identifier: <S07>
Description: Component Cooling System rate for flow to RHR heat exchanger.

69. EPM-4-B.7 001

Given the following events and conditions:

Unit 1 was conducting control rod drop tests during a plant startup at 2% reactor power when a complete loss of 'A' Train CCS occurred.

- Control room operators enter AOP-M.03 (Loss of Component Cooling Water)
- RCP Thrust Bearing temperature annunciator actuates.
- The operators manually trip the reactor but the trip breakers fail to open.
- Reactor power has increased to 5%
- Pressurizer pressure = 1930 psig

Which ONE (1) of the following statements correctly describes the proper procedural flow path for these conditions?

A. Remain in AOP-M.03, trip all RCPs and commence a reactor shutdown.

B. Implement FR-S.1 (Nuclear Power Generation/ATWS) concurrently with AOP-M.03.

C. Terminate AOP-M.03, enter E-0 (Reactor Trip or Safety Injection) and immediately transition to FR-S.1.

✓D. Enter E-0 and immediately transition to FR-S.1 while continuing on in AOP-M.03 as time and conditions permit.

A. Incorrect, per reference.
B. Incorrect, per reference.
C. Incorrect, per reference.
D. Correct, per reference.

K/A [CFR]: 2.4.5 [2.9/3.6] [41.10 43.5]

References: EPM-4

LP/Objectives: OPL271C266, b.7

History: Procedure bank

Level: Comprehension ✓

Comments: FHW 12/02 2.4.5 ✓

43.5

3.11.6 Use of AOPs Within the EOP Network

A. EOPs have priority over AOPs at all times, except when a reactor trip or safety injection has occurred in conjunction with an Appendix R fire (AOP-N.08), Control Room abandonment (AOP-C.04), or Loss of all ERCW capability (AOP-M.01).

B. AOP performance while in the EOP network is allowable under the following two circumstances: **[C.1]**

1. AOP performance is directed by EOPs in effect.

2. AOP performance is deemed necessary by the SM or US to address abnormal plant conditions NOT directly addressed by the EOPs but which have a significant impact on the ability of the EOPs to perform their function (e.g., loss of ERCW, CCS, off-site power, vital instrument power board, etc.) In this case, the following guidelines should be followed:

   a. Concurrent performance of the EOPs and the AOP should enhance, NOT degrade, the performance of EOPs in progress.

   b. Manpower resources are adequate to allow performing the EOPs and the AOP concurrently.

   c. The AOP should be performed using the "reader-doer" method so the procedure reader remains dedicated to the EOPs in progress, which are mitigative in nature. The SM may elect to deviate from this requirement when in ES-0.1.

   d. Certain AOPs may be required to be performed concurrently with the EOPs in order for the EOPs to function as intended; for example, loss of CCS, loss of ERCW, loss of air or vital power to equipment important to safety-- any of these could have a significant impact on the ability of the EOPs to achieve their goals.

   e. Upon transition to ES-0.1, the SM will designate the mitigating crew responsibilities as appropriate, based on the events in progress. For example, the procedure reader and OATC might perform an AOP while the CRO performs ES-0.1 as a reader-doer.

78. FISSION*PROD*POISON 052

Which of the following ~~occurrences~~ can cause ~~reactor~~ power range NIs to fluctuate between the top and bottom of the core when steam demand is constant? AND the mitigation strategy used.

A. steam generator level oscillations.

B. iodine spiking.

✓C. xenon oscillations.

D. inadvertent boron dilution.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A [CFR]: 015 A2.03 [3.2/3.5] [41.5 43.5]

References: OPL271C228

LP/Objective: OPL271C228 B.38

History: General Fundamental Bank

Level: Memory

Comments: FHW 12/02 015 A2.03 *

MISSING 2ND part of K/A

X. LESSON BODY: | INSTRUCTOR NOTES

M. Xenon Redistribution Following a Step Load Decrease from 100% Power to 50% Power

Objective B.38

1. Prior to the power reduction, the axial Xenon distribution is relatively flat due to the magnitude of the flux and the saturation of equilibrium Xenon.

   Show CP-68
   CP-69

2. When the step load decrease occurs, control rods move into the core to decrease power. This causes an overall reduction in the magnitude of the flux and also causes the axial power distribution to shift to the bottom of the core.

3. Initially there is no change in the axial Xenon or iodine distributions. However, as Xenon starts to buildin following the power reduction, the control rods start to step out to compensate for the negative reactivity insertion due to Xenon. This causes the axial power distribution to shift to the top of the core.

4. After 4 hours, control rods are almost fully withdrawn due to Xenon buildin. The axial Xenon distribution has not changed significantly but the absolute magnitude is increasing. The axial iodine distribution has not changed significantly but the absolute magnitude has decreased. This occurs because the axial power distribution has been constantly changing since the power reduction (shifting toward the top of the core due to rod motion).

5. After 8 hours, the overall Xenon concentration is decreasing causing control rods to insert to compensate. Flux is high in the top of the core for > 2 hours which is now causing Xenon to burnout in the top of the core and buildin in the bottom. Iodine is beginning to buildin in the top and decay out in the bottom.

6. After 12 hours, the iodine increase in the top of the core is causing Xenon to buildin in the top and the iodine decrease in the bottom is causing Xenon to decay out in the bottom. This coupled with rod insertion due to overall Xenon reduction causes the power distribution to begin to shift to the bottom of the core.

X. LESSON BODY: INSTRUCTOR NOTES

7. After 16 hours, the Xenon changes in the top and bottom of the core and the rod insertion cause flux to peak in the bottom of the core.

8. After 20 hours, flux has been relatively stable and peaked in the bottom of the core to cause iodine to begin to buildin in the bottom and decay out in the top. Xenon is now building in in the top and burning out in the bottom. This is causing flux to peak even higher in the bottom of the core.

9. After 24 hours, iodine is peaking in the bottom of the core while Xenon is peaking in the top. This is causing flux to peak even higher in the bottom of the core.

10. Beyond 24 hours, rod motion is not significant because Xenon has built in to > 75% of its equilibrium value. Flux will begin to shift back to the top of the core as Xenon and iodine once again begin to build and burn in opposite ends of the core. The result is an axial Xenon oscillation that may or may not converge.

N. Xenon Transients

1. Rod motion is the most significant contributor to axial Xenon oscillations although temperature changes also contribute. — Objective B.39

2. Rod insertion for whatever reason (runback or rod motion at steady state) will cause flux to shift to the bottom of the core (provided rods are initially above the midplane of the core). — Show CP-70

3. If rods are allowed to stay inserted for a significant period of time ($\geq$ 2 hours), Xenon will start building up in the top and burning out in the bottom.

4. Xenon buildup in the top and burn out in the bottom will cause axial flux to shift even further to the bottom of the core. This could potentially cause hot channel factor and DNB problems in certain areas of the core. If rod motion is occurring at the same time, this will cause the flux to be even higher in the bottom of the core.

101. OPL271C368.3 001

Unit One is taking actions to mitigate a loss of offsite power. Fifteen (15) seconds after 1A-A 6.9kv shutdown board was energized by the 1A-A D/G the Operator noticed the 1A-A CCP was **NOT** running and the 1A-A AFW pump was **NOT** running.

What should be the Operator's next action? if any?

✓A. Manually start the 1A-A CCP, then monitor the 1A-A AFW pump for auto start.

B. Manually start the 1A-A CCP and the 1A-A AFW pump.

C. Nothing, the 1A-A CCP and the 1A-A AFW pump load sequencers have **NOT** timed out for starting.

D. Nothing, the 1A-A CCP and 1A-A AFW pump sequencers function only with a black out and SI signal combined.

A. Correct, the CCP should have auto started and the AFW pump sequencer has not timed out for auto start.
B. Incorrect, the AFW pump sequencer has not timed out for auto start.
C. Incorrect, the CCP sequencer should have auto started.
C. Incorrect, the CCP and AFW sequencer will function for a black out only as well as for a black out and SI combined.

K/A {CFR} 056 AK3.01 [3.5/3.9] {41.5 41.10}
064K4.10 [3.5-4.0] {41.7}

References: AOP-P.1 appendix B

LP/Objective: OPL271C368 B.4

History: Develop1 Bank

Level: ~~Memory~~. H

Comments; FHW 12/02 056 AK3.01 ✓

is it wrong to start AFW pump ahead of sequencer?

APPENDIX B

LOSS OF OFFSITE POWER DIESEL GENERATOR LOAD SEQUENCE [C.2]

NOTE Diesel generator is rated at 4400kW continuous or 4800kW for 2 hours in 24-hour period.

EQUIPMENT NAME	LOADED	TIME IN SECONDS (1)
Miscellaneous loads	BO or SI with BO	0
CCP	BO or SI with BO	2
SI Pumps	SI with BO only	5
RHR Pumps	SI with BO only	10
ERCW Pumps	BO or SI with BO	15
AFW Pumps	BO or SI with BO	20
Thermal Barrier Booster Pumps	BO or SI with BO	20
CCS Pumps	BO or SI with BO	30
Pressurizer Heaters (A-A & B-B only)	BO only	90
CS Pump	Phase B with BO	180
Main Control Room AHU	Phase B with BO(3)	220(3)
Electric Board Room AHU	Phase B with BO(3)	240(3)

(1) Time is measured from the time of closing the breaker connecting the D/G to the power train.

(3) Time delay only if phase B isolation is present.

105. PI-B.10 001

Which ONE (1) of the following radiation detectors will initiate a Containment Ventilation Isolation Signal on high radiation level?

A. Lower Containment Particulate Detector.

✓B. Containment Purge Radiation Detector.

C. Containment High-Range Area Detector.

D. Spent Fuel Pit Continuous Air Monitor Detector.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A[CFR]:	029 A1.02 [3.4/3.4] [41.5]
Reference:	0-SO-30-3
LP/Objective:	OPL271C013 B.4
History:	System bank, old Bank Number PL-0280
Level:	Memory
Comments:	FHW 12/02 029 A1.02 ✓

Unit____________ Date____________

8.3 Containment Ventilation System Isolation Actuation

NOTE 1 If the containment vent system is not in service and any of the dampers associated with the system are open, a containment vent isolation signal will close them.

NOTE 2 The following signals will actuate a containment ventilation isolation:

A. Safety Injection Signal

B. Phase "A" Containment Isolation

C. Phase "B" Containment Isolation

D. High radiation detected by containment purge air exhaust radiation monitor RM-90-130 or RM-90-131.

[1] **WHEN** a CVI actuation has occurred by one of the above signals, **THEN**

DISPATCH an operator locally to stop the pumps on RM-90-106 and RM-90-112 to prevent the pumps from possibly burning up while running with the isolation valves closed. ________

108. PZR PRESS-B.9 006

Given the following plant conditions:

- Unit 2 is operating at 100% power.
- PZR level at 60% and both PZR spray valves in manual and closed.
- MIG is investigating erratic responses.
- A main turbine control failure results in a rapid load reduction.
- RCS temperature, PZR level, and PZR pressure rise rapidly.
- The RO stabilizes RCS pressure at 2300 psig by manually opening one spray valve.
- Pressure is held constant at 2300 psig for an extended period of time.
- The RO then observes that PZR level is 68%.
- Pressurizer pressure master pressure controller output has increased to 85%.
- PCV-68-340 and PCV-68-334 are closed.
- The "Pressurizer Level High / Low" alarm is **NOT** lit.
- Backup heaters are on.

Which ONE (1) of the following describes the status of the Pressurizer Pressure & Level Control systems?

✓A. Functioning properly. ~~with sprays in manual.~~

B. Malfunctioning because PCV-68-334 and 340 should be open.

C. Malfunctioning because "Pressurizer Level High\Low" should be in alarm.

D. Malfunctioning because the backup heaters should be de-energized.

A. Correct per reference.
B. Incorrect the 2335 psig bistable logic is not made.
C. Incorrect the alarm setpoint is 70% span increasing, 5 percent of span deviation below program level. 68% level is above program level but less than 70%.
D. Incorrect, the backup heaters should be on because PZR level is more than 5% of span above level program. For 100% power PZR program level is 60%.

108. PZR PRESS-B.9 006

K/A {CFR}: 027 G 2.1.7
2.1.31 [4.2/3.9] [45.12]

References: AOP-I.04 and 2-AR-M5-A (C-3), (E-4)

LP/Objectives: OPL271PZRLCS B.9
OPL271PZRPCS B.9

History: System bank

Level: Comprehension

Comments: FHW 12/02 2.1.31

Source	Setpoint
SER 1216	70% span increasing
LS-68-335D/E - High	5 percent of span
SER 1217	deviation below level
LS-68-339F/E - Low	program

LS-68-335D/E PRESSURIZER LEVEL HIGH-LOW

Probable Causes

1. RCS leak exceeding charging capacity.
2. Load transient or RCS temperature transient condition.
3. Charging and/or letdown flow mismatch.
4. Instrument malfunction for level or Tavg.

Corrective Actions

[1] **CHECK** pressurizer level (2-LI-68-339A, 335A, 320)

[2] **IF** level is high, **THEN**
ENSURE backup heaters ON.

[3] **ENSURE** level control system is attempting to return level to program with letdown and charging.

[4] **IF** level channel failed, **THEN**
GO TO AOP-I.04, Pressurizer Instrument Malfunction.

[5] **IF** RCS leakage is suspected, **THEN**
GO TO AOP-R.05, RCS Leak and Leak Source Identification.

[6] **IF** in MODE 4 or MODE 5 and a LOCA is identified, **THEN**
GO TO AOP-R.02, Shutdown LOCA (MODE 4, or 5).

[7] **EVALUATE** Technical Specifications 3.3.1, 3.3.2 and 3.4.6.2 as applicable.

References

45B655-05A,
47B601-68-45

Source	Setpoint	
SER 367 2-LS-68-339E/F	5% of span above level program	**LS-68-339E/F PRESSURIZER LEVEL HIGH BACKUP HTRS ON**

Probable Causes

1. Charging and/or letdown flow mismatch.
2. Instrument malfunction of level or Tavg.
3. Load transient condition.

Corrective Actions

[1] **CONFIRM** instrumentation by CHANNEL CHECK.

[2] **IF** instrument has failed, **THEN**
GO TO AOP-I.04, Pressurizer Instrument Malfunction.

[3] **IF** instrument has <u>not</u> failed, **THEN**
ENSURE level is returning to program 2-LR-68-339 with appropriate charging and letdown.

[4] **IF** RCS pressure ≥ 2265 psig, **THEN**
DEENERGIZE backup heater 2C. **[C.1]**

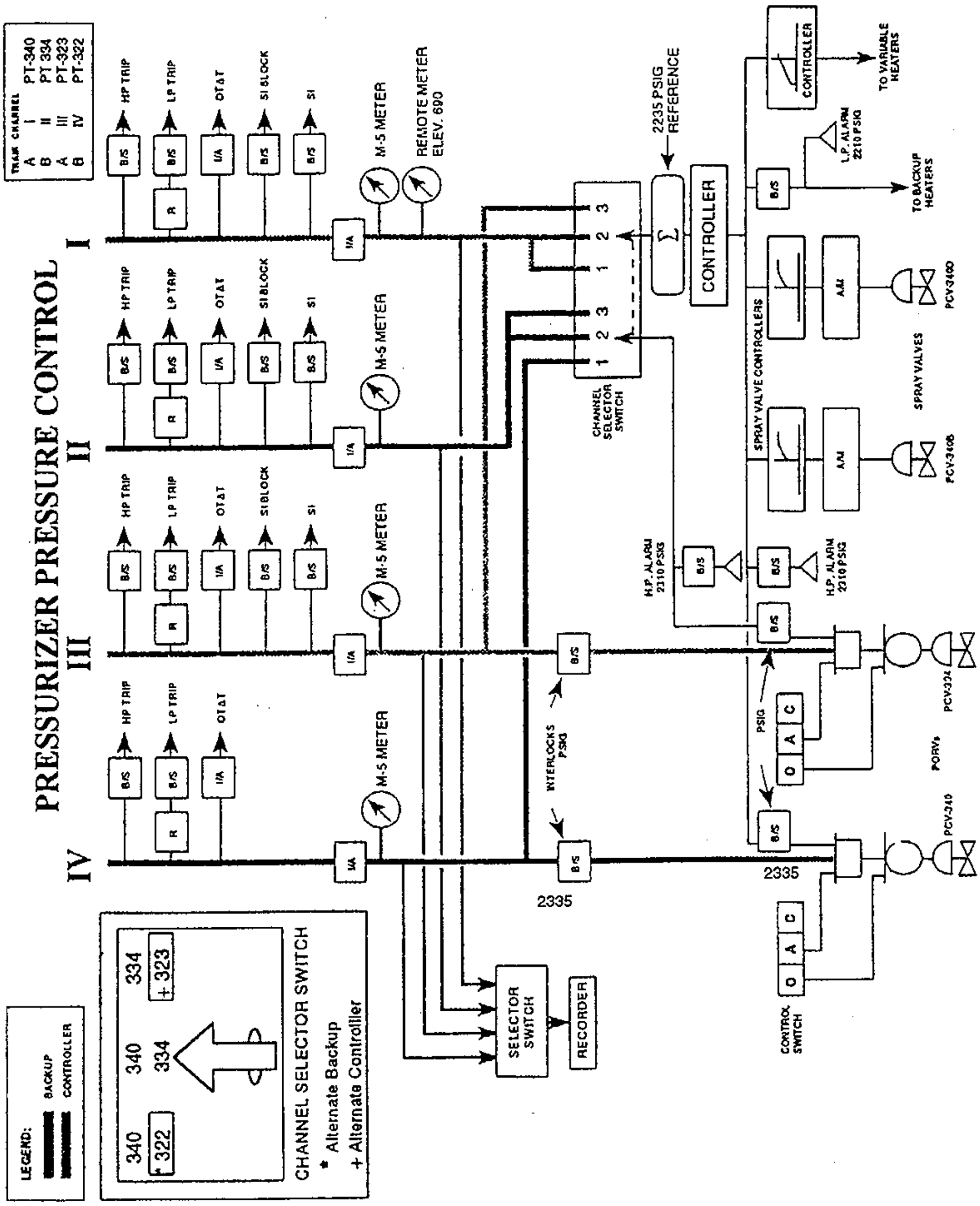
[5] **EVALUATE** Technical Specifications (3.3.1 and 3.3.2).

References

45N657-15,
45B655-05A,
47B601-68-45

APPENDIX H

PRESSURIZER PRESSURE CONTROL



110. RADWASTE-B.12 006

Which ONE ~~(1)~~ of the following explains why the Liquid Radwaste Processing System Monitor Tank is recirculated for one hour before being sampled?

A. Eliminates thermal stratification.

✓B. Provides homogeneous mixture for sampling.

C. Minimizes accumulation of radioactive material in tank.

D. Reduces deposition of chemical precipitates (boron) in tank.

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A [CFR]:	068 K4.01 [3.4/4.1] [41.7]
Reference:	ODCM section 6.1.1
LP/Objective:	OPL271LRW B.12
History:	System bank
Level:	Memory
Comments:	FHW 12/02 068 K4.01

Knowledge of design features or interlocks which provide ~~for~~ the following: safety and environmental precautions.

Table 2.2-1 RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
(Page 3 of 3) TABLE NOTATION

d A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by the method described in ODCM Section 6.1.1, to assure representative sampling.

e A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the continuous release.

f The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141. Ce-144 shall also be measured with an LLD of $5x10^{-6}$. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

g Releases from these tanks are continuously composited during releases. With the composite sampler or the sampler flow monitor inoperable, the sampling frequency shall be changed to require representative batch samples from each tank to be released to be taken prior to release and manually composited for these analyses.

h Applicable only during periods of primary to secondary leakage or the release of radioactivity as detected by the effluent radiation monitor provided the radiation monitor setpoint is set to alarm if activity in the stream exceeds a routine normal background, or compensatory requirements associated with applicable inoperable monitors are met.

117. REFUELING-B.1.G 001
Which ONE ~~(1)~~ of the following identifies the requirement for RHR cooling loops while in the refueling mode with no core alterations near the hot legs?

✓A. When > 23 feet in cavity only one RHR loop is required to be operable and in operation.

B. When < 23 feet in cavity only one RHR loop is required to be operable.

C. When < 23 feet in cavity two RHR loops are required to be operating.

D. When > 23 feet in cavity two RHR loops are required to be operable and one in operation.

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A: 2.2.26 [2.5/3.7] [43.5]

Reference: Tech Spec 3.9.8.2
Tech Spec 3.9.8.1

Objective: OPL271c269, b.1.g

Level: Memory

History: Procedure bank

Comments: FHW 12/02 2.2.26✓

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2000 gpm at least once per 12 hours. R138

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.* R16

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet. R217

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop. R16

R217

124. RVINT-B.8 001

Technical Specification 3.4.9, Pressure/Temperature Limits curves are for limiting Reactor Coolant System cooldown rates.

Which ONE ~~(1)~~ of the following describes the technical specification basis for using the composite curves for limiting reactor vessel cooldown?

A. The thermal gradients produced during cooldown produce compressive stresses at the inside of the reactor vessel wall.

B. The thermal gradients produced during cooldown produce tensile stresses at the outside of the reactor vessel wall.

✓C. The cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the reactor vessel temperature at the tip of the assumed flaw.

D. The cooldown procedure is based on measurement of reactor coolant pressure, whereas the limiting temperature is actually dependent on the reactor vessel stress at the tip of the assumed flaw.

A. Incorrect because the thermal gradients produced during cooldown produce tensile stresses at the inside of the reactor vessel wall.

B. Incorrect because the thermal gradients produced during cooldown produce compressive stresses at the outside of the reactor vessel wall.

C. Correct because there is no way to measure the reactor vessel temperature at the tip of the assumed flaw.

D. Incorrect because the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the reactor vessel temperature at the tip of the assumed flaw.

K/A: 2.2.25
039 K5.05 [2.7/3.1] [41.5]

Reference: Technical Specifications, page B 3/4 4-12

LP/Objective: OPL271RVINT, B.5

History: New question (Developed 7/17/98).

Level: Memory

Comments: FHW 12/02 039 K5.05✓ operational implications?

stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive

28. T/S0304.02 003

Unit One is starting up from Cold Shutdown. The 0-SI-OPS-000-004.0 procedure has acceptance criteria that requires, the primary and secondary Steam Generator metal temperatures shall be verified greater than 70°F on an hourly basis when RCS or Steam Generator pressures is greater than 200 psig and no RCP is in service.

This acceptance criteria will prevent

A. rapid depressurization of the RCS and subsequent injection of non-condensible gases upon RCP start.

B. subsequent reactivity excursion on RCP start.

C. pressurized thermal shock of the reactor vessel.

✓D. pressurized thermal shock of Steam Generators.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A [CFR]: 003 K4.02 [2.5/2.7] [41.7]

Reference: 0-GO-1 section 3.2.K
0-SI-OPS-000-004.0 Appendix A Table 1 acceptance criteria.
TS 3.7.2 and TS bases 3/4.7.2.

LP/Objective: OPL271C181 B.2

History: Develop one bank.

Level: Comprehension

Comments: Modified the question to SQN LCO 3.7.2 which has different criteria.
FHW 12/02 003 K4.02

What does this have to do with knowledge of RCP design features or interlocks which provide for cold water accidents?

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig. R16

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F. R16

MAR 25 1982

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 MAIN FEEDWATER ISOLATION, REGULATING, AND BYPASS VALVES

Isolation of the main feedwater (MFW) system is provided when required to mitigate the consequences of a steam line break, feedwater line break, excessive feedwater flow, and loss of normal feedwater (and station blackout) accident. Redundant isolation capability is provided on each feedwater line consisting of the feedwater isolation valve (MFIV) and the main feedwater regulating valve (MFRV) and its associated bypass valve. The safety function of these valves is fulfilled when closed or isolated by a closed manual isolation valve. Therefore, the feedwater isolation function may be considered OPERABLE if its respective valves are OPERABLE, if they are maintained in a closed and deactivated position, or if isolated by a closed manual valve. The 72-hour completion time to either restore, close, or isolate an inoperable valve takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring that would require isolation of the MFW flow paths during this time period. The 8-hour completion time for two inoperable valves in one flow path takes into account the potential for no redundant system to perform the required safety function and a reasonable duration to close or isolate the flow path. Although the steam generator can be isolated with the failure of two valves in parallel, the double failure could be an indication of a common mode failure and should be treated the same as the loss of the isolation function. The 7-day frequency to verify that an inoperable valve is closed or isolated is reasonable based on valve status indications available in the control room, and other administrative controls to ensure the valves are closed or isolated. R236

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 25°F and are sufficient to prevent brittle fracture. BR

Unit__________ Date____________

APPENDIX A

Page 3 of 3

[2] **IF** primary or secondary side pressures are greater than 200 psig, **THEN**

RECORD information in Table 1 once each hour until all S/G primary or secondary pressures are indicating less than 200 psig or a RCP is running. ☐

NOTE PI-1-33 should be used only for secondary pressure indication when the MSIV's are open, otherwise, any one indicator may be used and logged in the table. This Table may be copied as needed for additional hourly record.

TABLE 1

TIME							
SECONDARY	Pressure Indicator						
	Pressure (psig)						
	Appendix 'E' Data	S/G#1	S/G#1	S/G#1	S/G#1	S/G#1	S/G#1
		S/G#2	S/G#2	S/G#2	S/G#2	S/G#2	S/G#2
		S/G#3	S/G#3	S/G#3	S/G#3	S/G#3	S/G#3
		S/G#4	S/G#4	S/G#4	S/G#4	S/G#4	S/G#4
PRIMARY	Pressure Indicator						
	Pressure (psig)						
	Temperature Indicator						
	Temperature (°F)						
INITIALS							

ACCEPTANCE CRITERIA: The primary and secondary Steam Generator metal temperatures shall be verified >70°F on an hourly basis when RCS or Steam Generator pressures is >200 psig and no RCP is in service.

3.2 LIMITATIONS (Continued)

K. Prior to exceeding 200 psig in either the RCS or steam generator secondary side, perform 0-SI-OPS-000-004.0 on an hourly basis to verify temperatures greater than 70°F. Continue 0-SI-OPS-000-004.0 until pressure is less than 200 psig and/or an RCP has been started.

L. Greater than or equal to 200 psid is the #1 seal ΔP limit for RCP operation. However, when instrument inaccuracy is taken into account the ΔP limit has to be increased to 220 psid to maintain the required 200 psid.

M. When the RCS temperature is less than 350°F, both safety injection pumps and one centrifugal charging pump shall be made incapable of automatic injection into the RCS. This is necessary to minimize the potential for and the severity of low temperature overpressurization events (TS 3.4.12).

N. Whenever the RCS temperature is above 350°F, at least one train of control rod drive mechanism cooling fans must be in service to maintain shroud temperature less than 164°F.

O. Plant Computer is updating if the time indication on monitor is changing. **[C.4]**

P. After any significant change in charging flow, the RCP seal injection flows should be checked, and adjusted if necessary, to maintain a minimum of 6 gpm to each reactor coolant pump.

Q. The MSIVs should not be stroked more often than once per 10 minutes when the valve internals are dry (no steam in pipe).

R. The applicability of SR 4.4.6.3.c needs to be evaluated anytime an RCS pressure isolation valve listed in table 3.4-1 of Tech Specs has been actuated due to automatic or manual action or flow through the valve.

S. Credit can be taken for filled steam generators only if the RCS remains pressurized above 50 psig (TS 3.4.1.4). **[C.15]**

1. WGDS-B.12 001
   Given the following:

   - A waste gas decay tank release was in progress.
   - RM-90-118, Waste Gas Discharge Radiation Monitor, is in service.
   - 0-RA-90-118A WDS GAS EFF MON HIGH RAD annunicator alarms.
   - RCV-77-119, Gas Release Header Flow Control Valve, has tripped closed.
   - The Chemlab has performed a backup sample analysis on the waste gas decay tank.
   - A new release permit has been issued, RM-90-118 setpoint has been adjusted, and approval has been granted to resume the release.

   Which ONE of the following is (are) the MINIMUM action(s) to take for RCV-77-119 (Gas Release Header Flow Control Valve) in order to continue the waste gas release?

   A. Reset the valve locally and allow it to automatically re-open.

   ✓B. Take the valve controller to ZERO (0), then re-open the valve.

   C. Purge the release line with nitrogen until RM-90-118 returns to background, reset the valve locally, then re-open the valve.

   D. Place the PROCESS RADMON SYS BLOCK SW TRAIN "A" to the RM-90-118 position, take the controller to ZERO (0), then re-open the valve.

   A. Incorrect per reference.
   B. Correct per reference.
   C. Incorrect per reference.
   D. Incorrect per reference.
   Student must comprehend this situation requires a new procedure for the second release and begins at step one.

   K/A[CFR]: 071 A4.26 [3.1/3.9] [41.7]

   Reference: 0-SO-77-15 section 6.0 [15] and [22].

   Objective: OPL271GRW, B.12

   History: Modified from 97 Summer NRC Exam

   Level: Comprehension

   Comments: FHW 12/02 071 A4.26 ✓

Date ___________

6.0 NORMAL OPERATION (Continued)

NOTE The control switch for 0-FCV-77-245 is 0-HS-77-245, and is located in the Unit 1 690 pipe chase, approximately 40 feet east of the BIT approximately 5' from the floor.

[14] **ENSURE** **[0-FCV-77-245]** waste gas vent flow control valve is **OPEN** to the Shield Building vent indicated in step **[3]**. ___________

[15] **ADJUST** **[0-FIC-77-119]** (0-FIC-77-119, on panel 0-L-2A) release header flow control valve controller on panel 0-L-2A to the zero setpoint. ___________

[16] **OPEN** **[0-77-750]** release header filter outlet isolation valve. ___________

[17] **OPEN** **[0-77-749]** release header filter inlet isolation valve. ___________

[18] **RECORD** below the initial pressure of the gas decay tank to be released.

Tank ID (Letter)	Tank Pressure
______________	_________PSIG

____

[19] **ENSURE** the tank to be released is aligned to the "Release to Vent Header Alignment" in accordance with Attachment 3 Valve Checklist 0-77-15.03. ___________

Date __________

6.0 NORMAL OPERATION (Continued)

[20] **ENSURE** the following interlocks satisfied for 0-FCV-77-119:

[a] Selected unit's Shield Bldg. Exhaust flow > 3000 cfm (1, 2Y2210A). ☐

[b] High radiation on 0-RE-90-118 clear. ☐

[c] Instrument Malfunction on 0-RM-90-118 clear. ☐

[d] FIC-77-119 output signal at "ZERO". ☐

[21] **IF** any interlock(s) **NOT** satisfied, **THEN**

TAKE action to satisfy the interlock **OR INTIATE** WO for corrective action. WO#______________ __________

NOTE 1 When the gas release is initiated, the data requested on 0-SI-CEM-077-410.4, APP. B, should be recorded at initiation of the release and at 1 hour intervals.

NOTE 2 The following two steps will require two operators, one at panel 0-L-2A, and one at panel 0-L-473.

NOTE 3 The radiation control valve 0-RCV-77-119 will close on high radioactivity as indicated on 0-RE-90-118. If the high radiation alarm is present, then appropriate actions must be taken to clear the high radiation condition. The control signal must be zeroed before the valve will reopen.

[22] **OPEN SLOWLY [RCV-77-119]** (0-FIC-77-119 on panel 0-L-2A) gas release header flow control valve, **AND**

OBTAIN the release flow rate approved in 0-SI-CEM-077-410.4, as indicated on FE-77-230 flow indicator on panel 0-L-473 (inches H_2O). __________

56. D/G-B.10 002

Given the following plant conditions:

- Unit 1 & 2 are steady-state at 100% power
- 125V DC Vital Battery Board III is inadvertently deenergized

Which ONE ~~(1)~~ of the following describes the effect this has on the diesel generators?

A. All diesel generators except Diesel Generator 1A-A would auto start.

B. All diesel generators would auto start but all engine and generator trips on Diesel Generator 2A-A would be disabled.

C. Diesel Generator 1A-A would auto start and could only be shutdown using the EMERGENCY STOP pushbutton on 0-M-26 until control power was restored.

✓D. Diesel Generator 2A-A would auto start and could only be shutdown using the Local EMERGENCY STOP pushbutton until control power was restored.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.

K/A {CFR}: 064000K203 [3.2/3.6]
064 A3.06 [3.3/3.4] [41.7]

References: 1,2-45N767-2 & 5

LP/Objectives: OPL271 D/G, B.9

History: HLC 9809 Audit Exam

Level: Analysis

Comments: FHW 12/02 064 A3.06

K/A man

116. RDCNT-B.7.B 001

Unit 1 was operating at 60% power. Given the following events and conditions:

- RTA Breaker A is open for testing.
- Pressurizer pressure decreased to 1940 psig.
- The SSPS train "A" low PZR pressure trip logic relay failed to actuate.

What effect would this failure have on the function of the reactor protection system?

A. The reactor would **NOT** trip because the Train A logic relay would not remove power from the UV coil for ~~RTA~~. BYA.

B. The reactor would **NOT** trip because the Train B logic relay would not remove power from the UV coil for ~~RTA.~~ BYA.

✓C. The reactor would trip because the Train B logic relay would remove power from the UV coil for ~~RTB~~. BYA.

D. The reactor would trip because the Train B logic relay would remove power from the UV coil for ~~RTA~~. BYB.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A {CFR}: 012 K2.01 [3.3/3.7] [41.7]

References: 1,2-47W611-99-1

LP/Objectives: OPL271RPS, b.11

History: Systems bank

Level: Comprehension

Comments: FHW 12/02 012 ~~K2.01~~

K4.04
or
K4.01

RO/SRO/KA Cross Reference

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
001 AK1.03	AOP-C.01-B.1 002	B	C	A
001 K.301	001 K.301	B	A	N
001 K5.30	CONTROL*RODS 020	B	M	A
003 A1.03	RCP-B.12 002	B	M	A
004 A3.07	004 A3.07	B	C	N
005 A1.01	RHR-B.13.H 001	B	M	N
006 A1.07	ECCS-B.3 002	B	M	A
006 K6.19	006 K6.19	B	C	N
008 K4.02	CCS-B.9.A 001	B	A	A
009 EK1.01	INPO3 955	B	C	A
010 A2.02	010 A2.02	B	M	N
012 K6.07	012 K6.07	B	C	N
013 K4.01	013 K4.01	B	M	N
015 AK2.08	AOP-R.05-B.5 001	B	M	A
016 A3.02	RCS-TEMP-B.2 008	B	M	A
016 K1.01	INCORE-B.1.B 002	B	M	A
017 K6.01	INCORE-B.1.D 003	B	M	A
022 K2.01	022 K2.01	B	C	N
024 AA1.26	ES-0.1-B.1 002	B	C	A
025 AK2.05	FR-Z.1-B.2 001	B	C	A
025 K5.02	CTMT-B.11 001	B	C	A
027 AK1.02	OPL271C353.4 001	B	C	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
029 EK2.06	RPS-B.5.B 001	B	C	A
034 A1.02	034 A1.02	B	A	N
034 K4.02	FH-B.12.B 001	B	M	A
035 K5.01	035 K5.01	B	C	N
037 2.1.7	AOP-R.01-B.2 003	B	A	A
037 AA1.11	AOP-R.01-B.2 004	B	M	A
038 EK3.08	INPO2 976	B	C	A
039 K3.04	FW-B.5 001	B	A	A
040 AK1.01	INPO2 231	B	C	A
051 AK3.01	SDCS-B.12 023	B	C	A
058 AK3.01	AOP-P.02-B.4 002	B	C	A
059 A4.01	059 A4.01	B	C	N
059 K1.04	MFW 001	B	M	A
060 AK2.02	060 AK2.02	B	C	N
061 AK2.01	061 AK2.01	B	A	N
061 K2.01	061 K2.01	B	M	N
061 K6.01	061 K6.01	B	M	N
062 A4.03	INPO8 155	B	M	A
063 A2.01	063 A2.01	B	M	N
063 K4.04	FW-B.5.B 004	B	M	A
064 K2.02	064 K2.02	B	C	N
068 A3.02	RADWASTE-B.12 008	B	M	A
069 AA1.03	CTMT-B.5 004	B	C	A
071 A1.06	RMS-B.9 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
072 A1.01	RMS-B.2 004	B	M	A
074 EK1.01	FR-C.1-B.2 003	B	C	A
074 EK2.02	FR-C.1-B.2 011	B	A	A
075 K1.01	RCW-B.9 001	B	M	A
076 AK3.05	076 AK3.05	B	C	N
078 K1.01	AIR-B.12 002	B	C	A
079 K4.01	AIR-B.5 014	B	C	A
103 A3.01	103 A3.01	B	M	N
2.1.24	2.1.24	B	A	N
2.2.11	CTMT-B.11 004	B	M	A
2.2.4	2.2.4	B	A	N
2.3.11	ODCM-B.5 001	B	C	A
2.3.9	CTMT PURGE-B.4 001	B	M	A
2.4.39	REP-B.1.D 002	B	M	A
2.4.4	AOP-R.02-B.2 001	B	C	A
E01 EA1.1	ES-0.0-B.3 001	B	C	A
E03 EA1.2	ES-1.2-B.2 006	B	C	A
E03 EK1.3	ES-1.2-B.2 004	B	M	A
E04 EK2.2	ECA-1.2-B.1 002	B	M	A
E05 EA1.3	FR-H.1-B.3 004	B	C	A
E08 EK3.2	FR-P.1 001	B	C	A
E09 EA1.1	OPL271C382.4 001	B	A	A
E10 EK3.1	ES-0.2-B.3 003	B	C	A
E14 EK1.3	ECCS-B.2 002	B	M	A
E15 EK1.1	FR-Z.2-B.2 001	B	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
003 K4.02	T/S0304.02 003	R	C	A
005 AA2.01	005 AA2.01	R	C	N
008 A2.04	008 A2.04	R	C	N
012 K2.01	RDCNT-B.7.B 001	R	C	A
015 A2.03	FISSION*PROD*POISON 052	R	M	A
022 2.4.27	022 2.4.27	R	A	N
025 2.2.13	025 2.2.13	R	M	N
029 A1.02	PI-B.10 001	R	M	A
039 K5.05	RVINT-B.8 001	R	M	A
056 A2.04	056 A2.04	R	C	N
056 AK3.01	OPL271C368.3 001	R	M	A
056 K1.03	056 K1.03	R	M	N
059 AA2.05	059 AA2.05	R	M	N
064 A3.06	D/G-B.10 002	R	A	A
067 AA2.17	AOP-C.04 001	R	M	A
068 AK2.07	D/G-B.6 012	R	M	A
068 K4.01	RADWASTE-B.12 006	R	M	A
071 A4.26	WGDS-B.12 001	R	C	A
078 K3.03	078 K3.03	R	C	N
079 2.1.1	079 2.1.1	R	M	N
086 K3.01	086 K3.01	R	C	N
2.1.16	2.1.16	R	M	N
2.1.31	PZR PRESS-B.9 006	R	C	A
2.2.26	REFUELING-B.1.G 001	R	M	A
2.3.10	2.3.10	R	C	N

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.4.20	AOP-M.01-B.3 002	R	M	A
2.4.5	EPM-4-B.7 001	R	C	A
E01 2.2.25	E01 2.2.25	R	C	N
E15 EK2.2	E15 EK2.2	R	C	N

003 DROP ROD 2.4.4	AOP-C.01-B.5 011	S	M	A
004 2.4.1	OPL271C367.1 002	S	C	A
007 2.1.10	PRT-B.7 004	S	M	A
010 2.4.47	AOP-I.04-B.2 001	S	A	A
011 2.1.6	OPL271C367.1 003	S	C	A
013 2.4.47	E-0-B.6 003	S	M	A
015 AA2.07	AOP-R.04-B.5 001	S	A	A
022 AA2.01	AOP-R.05-B.2 001	S	A	A
025 AA2.06	RHR-B.12.A 003	S	M	A
027 AA2.04	T.S-2.1-B.2 001	S	A	A
040 AA2.01	OPL271C379.3 001	S	A	A
056 AA2.18	E-0-B.3.A 007	S	C	A
069 2.1.14	069 2.1.14	S	C	N
2.1.10	OPL271C458.1 001	S	M	A
2.1.22	GO-2-B.1 003	S	M	A
2.1.7	PZR LEVEL-B.14 003	S	A	A
2.2.25	OPL271C180.3 001	S	M	A
2.2.9	SPP-9.5 001	S	M	A
2.3.1	RCI-15 001	S	M	A

K/A ID	Question ID	RO/SRO S=SRO R=RO B=BOTH	M=Memory C=Comph A=Analysis	Type A=As Is M=Modified N=New
2.3.2	RADIATION 002	S	M	A
2.3.3	AOP-C.04-B.5 008	S	M	A
2.4.33	OPDP-4 002	S	M	A
2.4.45	AOP-M.03-B.1 002	S	C	A
E01 EA2.1	ES-0.0-B.5 002	S	C	A
E04 EA2.1	ECA-1.2-B.2 001	S	C	A
E09 EA2.1	ES-0.2-B.3 001	S	C	A
E12 2.4.16	ECA-2.1-B.1 004	S	A	A
E15 EA2.1	E15 EA2.1	S	M	N
E16 2.4.41	E16 2.4.41	S	C	N

27. T.S-2.1-B.2 001

This question has reference material attached.

Unit 2 was operating at 60% power when an event occurred. All 3 pressurizer safety valves lifted. T-ave peaked at 680°F and the RCS pressure transient reached 2675 psig.

The ~~unit was manually t~~ripped and RCS temperature and pressure rapidly decreased.

Which ONE (1) of the following describes the Tech Spec action applicable to this transient (prior to rods tripping)?

A. Reduce the RCS pressure to within its limit within 5 minutes.

B. Reduce the RCS pressure to within its limit within 1 hour.

C. No action is required because RCS pressure never exceeded 2735 psig.

✓D. Be in Hot Standby within 1 hour.

A. Incorrect because initial conditions are mode 1.
B. Incorrect because RCS pressure never exceeded 2735 psig.
C. Incorrect TS 2.1.1 was exceeded.
d. Correct per references.

K/A {CFR}:	027 AA2.04 [3.7/4.3] [43.5]
References:	Tech Spec Safety Limit 2.1.1 and Figure 2.1-1.
LP/Objectives:	OPLI271C075, B.2
History:	Tech Spec bank
Level:	Analysis

Comments:
FHW 12/02 027 AA2.04 ; Modified the stem and distractors for analysis. **Provide safety limits figure 2.1-1. DO NOT provide safety limit 2.1.1 since this is a required memory one hour action.**

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1. R45

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

Revision 23

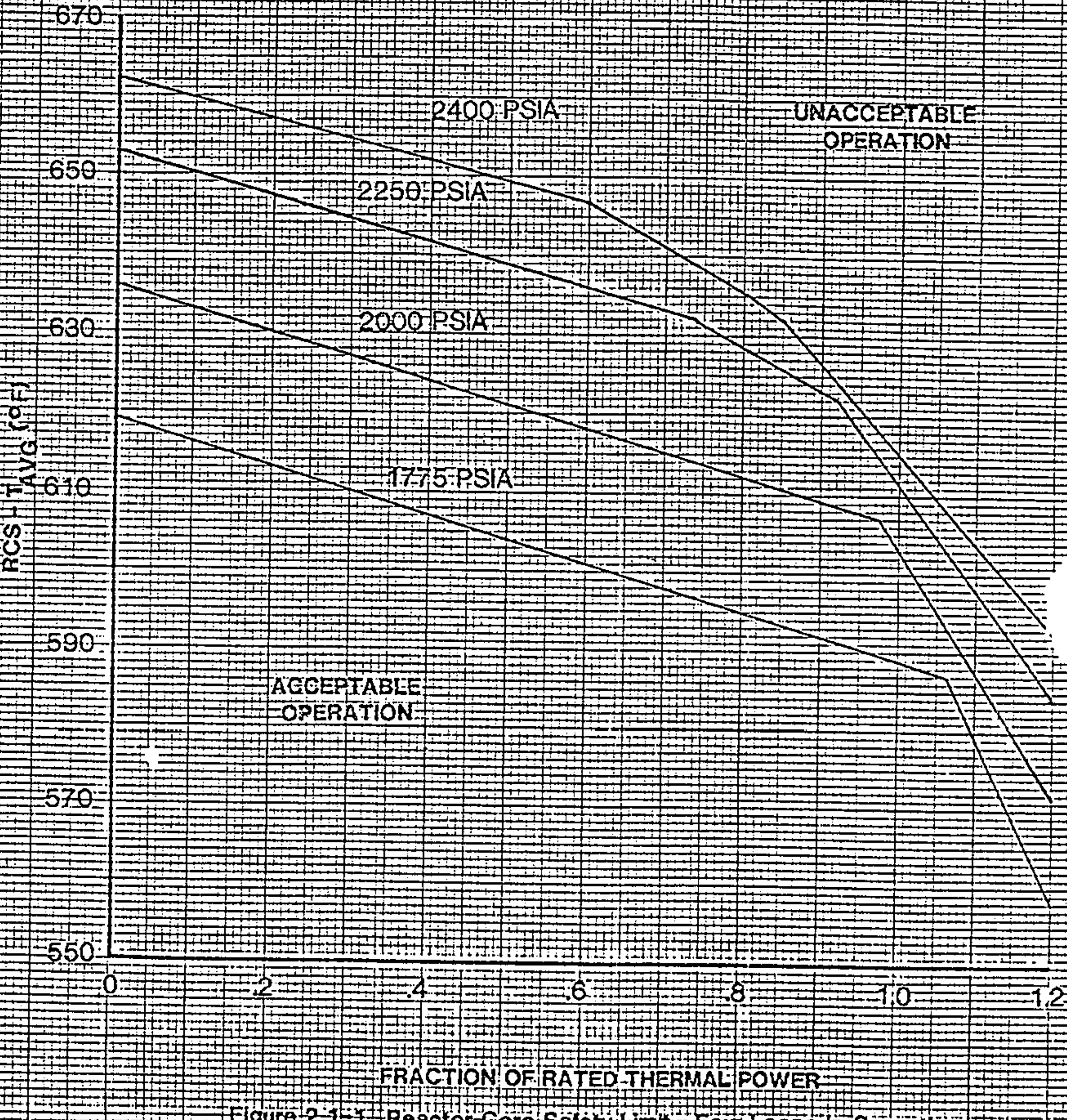


Figure 2.1-1. Reactor Core Safety Limit – Four Loops In Operation

26. SPP-9.5 001

Which ONE (1) of the following states responsibilities of the designated Senior Reactor Operator for normal installation of Temporary Alterations (TA) on quality related equipment?

A. Determine the need for environmental evaluation by the Environmental Section and reactivity management evaluation by Reactor Engineering.

B. Verify all required reviews and documentation have been completed, prepare the TA Tags and place the TA tags after the TA is installed.

✓C. Review the TACF to ensure required reviews AND engineering 50.59 reviews have been completed and documentation have been completed and that the TA can be performed without adverse effect on plant operation.

D. Perform a 10CFR50.59 evaluation on the TA to ensure it won't have an adverse affect on plant operation and review the TA to determine if training is needed prior to installation.

A. Incorrect - This is done by the Environmental Section
B. Incorrect - Performed by System or Design Engineer
C. Correct - per SPP 9.5 "Temporary Alterations"
D. Incorrect - Done by System or Design Engineer

K/A {CFR}:	2.2.9	[2.0/3.3]	[43.3]
References:	SPP-9.5		
LP/Objectives:	OPL271OPSMGMTL B.11		
History:	Procedure bank		
Level:	Memory		
Comments:	FHW 12/02 2.2.9		

1. If a TACF is no longer needed, prior to installation, the requester voids the TACF number in the site TACF Log Book.

2. The requester provides the original TACF to the TACF Coordinator for canceling.

3.3 Installation

NOTE Refer to Appendix C for material requirements.

A. Precautions and Limitations

   1. Ensure lifted leads are suitably insulated and jumpers securely attached.

   2. Verify the affected component is properly identified and in the correct circuit.

   3. De-energize electrical circuits prior to the installation of jumpers or the removal of leads, whenever possible.

   4. Ensure Category 1 drawings are updated and in the Control Room.

   5. Ensure secondary drawings are listed and referenced in Curator.

   6. Ensure materials (cables, jumpers, etc.) are equal to or better than the quality engineered into the host component or system.

   7. If the original TACF is removed, a copy of the TACF should replace the original TACF while it is removed.

B. Concerning installation, the SM or designee is responsible for the following:

   1. Review the TACF to ensure required reviews and documentation have been completed and that the TA can be performed without adverse effect on plant operation.

   2. Authorize installation of the TA and maintain control of the approved TACF. Sign and record date.

C. The System Engineer shall perform the following:

   1. Ensure the WID number is recorded.

   2. Verify all post installation testing has been successfully completed, if required. Sign and record date.

   3. Verify TA installation and TACF tags, if required, have been properly placed. Sign and record date.

   4. Forward a copy of the TACF to the TACF Coordinator. Sign and record date.

119. RHR-B.12.A 003

Which ONE ~~(1)~~ of the following correctly describes the operation of the RCS to RHR System Supply Valves FCV-74-1 and FCV-74-2?

A. RCS pressure > 687 psig causes auto-closure.

B. RWST suction valve 63-1 SHUT prevents opening

✓C. RCS pressure ≥ 380 psig prevents opening

D. RCS temperature of > 350°F causes auto-closure

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A[CFR]: 025 AA2.06 [3.2/3.4] [43.5]

Reference: 0-SO-74-1

LP/Objective: OPL271RHR B.9

History: System bank.

Level: Memory

Comments: FHW 12/02 025 AA2.06

Need to meet K/A statement on Loss of RHR i.e. overpressure protection when RHR is Lost.

Unit ____________ Date__________

5.6 Startup of RHR System for Normal Cooldown Mode (Continued)

NOTE **Refer to Tech Spec LCO 3.5.3.**

[8] **PLACE** both RHR pumps in PULL TO LOCK. ________

[9] **CLOSE** the following:

VALVE NO.	FUNCTION	INITIALS
FCV-63-1	RHR Suction from RWST	________
FCV-63-93	RHR Pump A-A Discharge to Loops 2 and 3 Cold Leg	________
FCV-63-94	RHR Pump B-B Discharge to Loops 1 and 4 Cold Leg	________

[10] **ENSURE** the following are **CLOSED**:

VALVE NO.	FUNCTION	INITIALS
FCV-63-8	RHR Hx A to CVCS Chg Pumps	________
FCV-63-11	RHR Hx B to SIS Pumps	________
HCV-74-34	RHR to RWST Return Isol	________

CAUTION **Opening FCV-74-1 and 2 with the pressurizer solid may cause a large drop in RCS pressure. Do not continue this section if the pressurizer is solid with RCPs running.**

NOTE 1 Pressure interlocks prevent FCV-74-1 and FCV-74-2 from being opened until RCS pressure is < 380 psig.

NOTE 2 FCV-74-1 and FCV-74-2 are administratively controlled at motor control panel. An M-5 key will be needed for unlocking breakers.

[11] **UNLOCK** and **CLOSE** breaker for **[<u>FCV-74-1</u>]** in Compt 6C2 on 480V Rx MOV Bd A1-A. ________

[12] **UNLOCK** and **CLOSE** breaker for **[<u>FCV-74-2</u>]** in Compt 14B on 480V Rx MOV Bd B1-B. ________

112. RCI-15 001

A pipe elbow on the spent resin transfer line in the El 690 Pipe Chase is producing a 500 mr/hr field at 30 centimeters from the elbow.

Which ONE (1) of the following identifies the proper posting requirement for the area surrounding the pipe elbow?

✓A. CAUTION, HIGH RADIATION AREA

B. DANGER, AIRBORNE RADIATION AREA

C. DANGER, ~~POTENTIAL VE~~RY HIGH RADIATION AREA

D. GRAVE DANGER, VERY HIGH RADIATION AREA

A. Correct per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A {CFR}: 2.3.1 [2.6/3.0] [41.12/43.4]

References: RCI-15 R12 section 6.8.

LP/Objectives: OPL271C260 B.3

History: HLC 9809 Audit Exam

Level: Memory

Comments: FHW 12/02 2.3.1; modified stem for a correct answer.

6.0 REQUIREMENTS (Continued)

6.7 Radiation Area

A. A Radiation Area is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (5 mrem) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

B. Each Radiation Area shall be posted with a conspicuous sign or signs bearing the radiation symbol and the words **CAUTION - RADIATION AREA**.

6.8 High Radiation Area

A. A High Radiation Area is an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 mrem) in one hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

B. Each High Radiation Area shall be posted with a conspicuous sign or signs bearing the radiation symbol and the words **CAUTION - HIGH RADIATION AREA** or **DANGER - HIGH RADIATION AREA**.

C. In lieu of the control devices or alarm signals required by **10CFR20**, each High Radiation Area in which the intensity of radiation is ≥ 100 mrem/hr at 30 cm, but < 1,000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a High Radiation Area and entrance shall be controlled by requiring issuance and use of an RWP. Any individual or group of individuals permitted entrance to such areas shall be provided with or accompanied by one or more of the following: **[C.2]**

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

109. RADIATION 002
Which ONE ~~(1)~~ of the following statements is correct concerning the SQN ALARA program?

✓A. The SQN Plant Manager must approve all lower containment entries inside the polar crane wall when the unit is in Mode 1.

B. The SQN Site Vice President must approve all lower containment entries inside the polar crane wall when the unit is in Mode 1 or 2.

C. The SQN Plant Manager must approve all lower containment entries inside the polar crane wall when the unit is in Mode 1 or 2.

D. The SQN Site Vice President must approve all lower containment entries inside the polar crane wall when the unit is in Mode 1.

A. Correct - As described in RCI-10 ALARA Program
B. Incorrect - Plant Manager must approve entry only in MODE 1.
C. Incorrect - Plant Manager must approve entry only in MODE 1.
D. Incorrect - Plant Manager must approve entry only in MODE 1.

K/A {CFR}:	2.3.2 [2.5/2.9] [41.12 43.4]
References:	RCI-10 R27 section 5.0.
LP/Objectives:	OPL271C260 B.9
History:	Admin bank
Level:	Memory
Comments:	FHW 12/02 2.3.2

Not SRO only Question

5.0 GENERAL (Continued)

M. All identified rework items shall be documented in the post job review of the **ALARA Planning Report** and clearly indicated as a rework item. Rework is defined as the repeat performance of a work activity because of improper performance of work, which could include but is not limited to, poor workmanship, re-machining, reassembling, poor engineering design or inadequate planning. **[C.2]**

N. The Plant Manager shall approve all Lower Containment entries inside the polar crane wall during Mode 1 operations. Approval will be documented on a form similar to that provided in **Attachment 05, Plant Manager's Approval for Lower Containment Entries Inside the Polar Crane Wall at Power**.

O. The RAD/CHEM Manager, or designee, shall approve all Containment Building entries during periods which are outside the pre-determined Containment Building entry schedule. Approvals shall be documented on an **Attachment 06, Containment Building Entry Request/Authorization**.

P. Individuals preparing work planning documents will ensure that the proposed work is evaluated for potential radiological impacts. The Radiological Impact Evaluator reviews the control document to determine if such impacts exist using **Appendix C**. Specific RADCON steps should be included in the applicable work plans or work orders for the job.

6.0 REQUIREMENTS

6.1 ALARA Pre-job Planning Criteria

Pre-job planning shall be carried out to identify and to minimize the hazards associated with jobs involving potential significant radiological hazards identified for each job where any of the following conditions occur:

A. The whole body (30 cm) dose rate is ≥ 1 rem/hr, and exposures are estimated to be greater than 50 mrem to any individual.

B. The collective total effective dose equivalent (TEDE) is expected to exceed 1 person-rem.

C. The collective extremity dose is expected to exceed 10 rem.

D. All entries into areas posted as a Very High Radiation Area.

E. At the discretion of the ALARA Coordinator.

107. PZR LEVEL-B.14 003•

Given the following plant conditions:

- Reactor power is 60%
- Pressurizer pressure is 2240 psig
- Charging flow is being controlled in MANUAL
- All other controls are in AUTOMATIC
- The backup heaters just ENERGIZED

Which ONE (1) of the following would be the indicated pressurizer level?

A. 41%

B. 43%

C. 46%

✓D. 51%

46% is the programmed PZR level for 60% power, +5% above program is the level which energizes the backup heaters.

K/A: 2.1.7 [3.7/4.4] [43.5]

Reference: TI-28
1-AR-M5-A (E-4)

Objective: OPL271C019, B.6

History: System bank

Level: Analysis

Comments: FHW 12/02 2.1.7 ✓

CONTROLS

TAVG

547 °F - 578.2 °F
(0 - 100% power)

PZR LEVEL

24.7% - 60%
(547 °F - 578.2 °F T_{avg} [Auct. Hi T_{avg}])

FEEDPUMP SPEED CONTROL

80 - 195 psid, (0 - 20 %, constant 80 psig)
(20 - 100% total steam flow)

S/G LEVEL

33 - 44% (0 - 20% turbine load)
44% (20 - 100% turbine load)

ROD CONTROL

Auto: 8 spm; 1.5°F → 3.0°F error
8 spm - 72 spm; 3°F → 5°F error

Manual: 48 spm

Bank Select: 48 spm control rods
64 spm shutdown rods

FP Rod Insertion Limit: 182 steps on "D" Bank

FIRST-STAGE IMPULSE PRESSURE

0 - 628 psia
0 - 100% power

STEAM DUMPS

Blocking:

Steam Dump Bypass Interlock Switches (M-4) in OFF

Condenser not available (absence of C-9)

Lo-Lo T_{avg} (P-12) locks out all steam dumps; interlock can be bypassed for 3 cooldown valves using the two bypass interlock switches on M-4.

Arming:

Load Rejection: 10% load decrease in a 2 minute time constant as sensed by PT-1-72 (C-7).

Reactor Trip: P-4 from 'A' Tr. Rx Trip Breakers ('B' Tr. P-4 places Rx Trip Controller I/S.)

Mode Selector Switch (M-4) in STEAM PRESSURE

Opening:

Tave Mode ⇒

Load Reject: Tave - Tref (PT-1-73)
(2°F → 18°F = 0 → 100% open)
Trip Open - ½ @ 10°F; ½ @ 18°F

Reactor Trip: Tave - 552°F (Fixed Reference Signal)
(0°F → 50°F = 0 → 100% open)
Trip Open - ½ @ 25°F; ½ @ 50°F

Pressure Mode: Auto = steam pressure - setpoint
Manual = Operator Controlled

Source	**Setpoint**	
SER 367 1-LS-68-339E/F	5% of span above level program	**LS-68-339E/F PRESSURIZER LEVEL HIGH BACKUP HTRS ON**

Probable Causes

1. Charging and/or letdown flow mismatch.
2. Instrument malfunction of level or Tavg.
3. Load transient condition.

Corrective Actions

[1] **CONFIRM** instrumentation by CHANNEL CHECK

[2] **IF** instrument has failed, **THEN**
GO TO AOP-I.04, Pressurizer Instrument Malfunction.

[3] **IF** instrument has not failed, **THEN**
ENSURE level is returning to program 1-LR-68-339 with appropriate charging and letdown.

[4] **IF** RCS pressure $\geq$ 2265 psig, **THEN**
DEENERGIZE backup heater 1C. **[C.1]**

[5] **EVALUATE** Technical Specifications (3.3.1 and 3.3.2).

References

45B655-05A-0,
45N657-15,
47B601-68-45

106. PRT-B.7 004

Unit 2 was operating at 90% after a start-up from a refueling outage. A PORV is found to be leaking and the PORV block valve was shut. Given the following PRT conditions:

- Level - 80%
- Pressure - 6 psig
- Temperature - 145°F

What action is required to restore normal operating conditions to the PRT?

A. Increase N2 ~~Blanket~~ BLANKET PRT pressure to 8 psig.

B. Vent/purge the PRT to the waste gas system.

✓C. Initiate cooling of the PRT.

D. Lower the PRT level .

A. Incorrect, increasing to 8 psig would bring in the high pressure alarm per reference.
B. Incorrect, venting is not required at 6 psig per reference.
C. Correct, alarm setpoint is 132° F per reference.
D. Incorrect, the level has not reached high alarm level of 88% per reference.

K/A[CFR]: 007 2.1.10 [2.7/3.9] [43.1]

References: 2-SO-68-5
2-AR-M5-A, (C-1)

LP/Objectives: OPL271PRT B.8

History: System bank

Level: Memory

Comments: FHW 12/02 007 2.1.10; modified "A" distractor to remove similar conditions as "B" distractor.

Source	Setpoint	
SER 1213 2-TS-68-309	132°F increasing	**TS-68-309 PRESSURIZER RELIEF TANK TEMP HIGH**

Probable Causes

1. Pressurizer safety valve or power relief valve open or leaking through.
2. Other relief valves (CVCS and RHR) or valve packing leakoff lines connected to PRT header have flow into the PRT.
3. Rx vessel head vent system leaking through.
4. Instrument malfunction.

Corrective Actions

[1] **CHECK** pressurizer relief tank temperature (2-TI-68-309).

[2] **CHECK** temperature on safety valve and power relief valve line.

[3] **VERIFY** vessel head vent isolation to PRT closed (2-HS-68-394 and 2-HS-68-395).

[4] **ADJUST** pressurizer relief tank temperature using 2-SO-68-5, *Pressurizer Relief Tank.*

[5] **IF** in MODE 4 or MODE 5 and a LOCA is identified, **THEN**
GO TO AOP-R.02, Shutdown LOCA (MODE 4, or 5).

[6] **IF** a small RCS leak is indicated, **THEN**
GO TO AOP-R.05, RCS Leak and Leak Source Identification.

[7] **EVALUATE** TS 3.4.6.2.

References

45B655-05A,
47B601-68-34,

2.2 Developmental References (Continued)

D. TVA Drawing
   1. 47W813-1
   2. 47W819-1
   3. 47W830-1
   4. 47W830-6

E. FSAR
   1. Section 5.5

3.0 PRECAUTIONS AND LIMITATIONS

A. During normal operation, PRT water temperature should not exceed 120°F.

B. During normal operation, maintaining approximately 3 to 6 psig N_2 gas blanket on the PRT will prevent the formation of explosive hydrogen-oxygen mixtures.

C. The PRT concentration of oxygen shall be limited to less than of equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

D. Over filling the PRT to solid water condition during oxygen reduction per Section 8.8 may result in failure of the PRT rupture disc.

E. The PRT pressure should be maintained < 7.5 psig during normal operation. (Except during the performance of section 8.9.)

F. The PRT rupture discs are rated at 85 psig.

G. During normal operation, the level in the PRT should be maintained at 70%. If the level increases to 88% then decreasing level to 70% is necessary. If the level decreases to 55% then increasing level to 70% is needed when the PRT is required to be operable.

H. Completely draining the PRT may result in gas binding the RCDT pumps.

I. Water intrusion into the waste gas vent header is possible during PRT venting operations with PRT level high. This could affect RCP seal leakoff flows and the vent capability of tanks which vent to waste gas vent header.

104. OPL271C458.1 001

Per SPP-10.0, Plant Operatons, under which one of the following conditions can an individual without a license manipulate controls that directly affect reactivity.?

A. Any ~~A~~ trainee under the direction and in presence of an ~~inactive~~ operative licensed Senior Reactor Operator.

B. When the individual has taken the NRC License Examination and is awaiting results for confirmation.

✓C. ~~A~~ licensed operator trainee under the direction and in presence of an active licensed Reactor Operator.

D. When the licensed Reactor Operator requires assistance and asks for any individual's help.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
d. Incorrect per reference.

K/A[CFR]: 2.1.10 [2.7/3.9] [43.1]

Reference: SPP-10.0 R2

LP/Objectives: OPL271OPSMGMTL B.21

History: Procedure bank, old Bank Number B-0743

Level: Memory

Comments: FHW 12/02 2.1.10: Modified stem and distractors to conform to requirements of SPP-10.0.

3.2 **Surveillance Areas for Operators**

The UO (unit operator) at the controls shall not leave the control room surveillance area (as shown in Appendix C) for any nonemergency reason without obtaining a qualified relief. In the event of an emergency affecting the safety of operations, the UO at the controls may momentarily be absent from the surveillance area in order to verify the receipt of an annunciator alarm or initiate corrective action provided the operator remains within the confines of the Control Room. The restrictions set forth for a Unit's surveillance area are not applicable for that unit when the reactor vessel is defueled.

3.3 **Personnel Recall**

The SM has the authority to call out all required personnel, regardless of discipline.

3.4 **Authority to Operate Equipment**

The overall operation of the plant shall be directed by the SM. The UO, Unit Supervisor (US), and the SM should be aware of all activities affecting plant equipment.

During emergencies, operators may take necessary immediate actions required to ensure personnel and plant safety, without prior approval; however, appropriate supervisors should be promptly informed of these actions.

Only licensed operators or individuals authorized by approved procedures should operate control room equipment. Only licensed operators are permitted to manipulate the controls that directly affect the reactivity or power level of a reactor except for training purposes. A trainee may manipulate controls that directly affect the reactivity or power level of a reactor only under the direction and in the presence of an active licensed operator. Trainees operating other plant equipment not affecting reactivity or power level shall be under the supervision of an individual qualified to perform the operations.

3.5 **Potentially Distractive Reading Material and Devices**

Reading material that does not relate to plant operation and entertainment devices such as radios, televisions, tape players, and computer games are prohibited from use by on-duty Operations personnel.

Unauthorized written material and entertainment devices shall not be brought to work stations. Unauthorized written material is that which is not job related or has not been issued by TVA. Judgment should be used to ensure the plant personnel's' primary duties are not compromised.

3.6 **Control Room Activities**

A. Potentially distracting activities in the control room and other watchstations are prohibited (e.g., radios, TV, games, horseplay, and hobbies).

B. Necessary plant-related technical/administrative control room business must be conducted at a location and in such a manner that neither licensed Unit Operator attentiveness nor the professional atmosphere will be compromised.

C. Access to the "surveillance area" (reference Appendix C) will be restricted and limited to official business only. Access for nonwork-related reasons will be prohibited. Permission to enter the "surveillance area" for personnel other than the Operations shift complement must be obtained from the UO, US or the SM.

100. OPL271C367.1 003

Unit 1 is at 100% RTP when a small RCS leak develops. The operating crew takes the necessary actions to stabilize PZR level. Subsequently to these actions the crew attempts to determine the leakage source. After isolating letdown, the crew observes the following parameters/indications:

Letdown flow 0 gpm
Cntmt pressure approximately 0.2 and decreasing
RCS pressure approximately 2235
Pzr level approximately 65% and increasing
Charging flow 40 gpm
Cntmt radiation monitors decreasing

Which ONE of the following actions should the SRO direct the operating crew perform next?

A. Proceed to Cold Shutdown per AOP-C.03 .

✓B. Place excess letdown inservice and adjust charging flow.

C. Immediately trip the reactor.

D. Decrease turbine load to approximately 50% at 2%/min.

A. Incorrect, AOP-C.03 Emergency Shutdown is entered for conditions that require a rapid shutdown without a reactor trip. Since pressurizer level is stabilized and the other initial conditions indicated the RCS leak has been isolated a rapid shutdown is not required per AOP-R.05.

B. Correct, the RCS leak was determined to be on normal letdown. Therefore to maintain pressurizer level and RCS chemistry (chemical feed and boration) excess letdown is required.

C. Incorrect, the initial conditions do not require a reactor trip.

D. Incorrect, a load reduction of 2%/min is within the guidelines of AOP-C.03, Emergency Shutdown which is not required since the RCS leak is isolated.

The student must understand from the initial conditions the RCS leak was on the letdown line and has been isolated. In order to maintain pressurizer level and chemistry excess letdown is required.

100. OPL271C367.1 003

K/A[CFR]: 011 2.1.6 [2.1/4.3] [43.5]

Reference: AOP-R.05
EA-62-3

LP/Objectives: OPL271C367 B.2

History: Procedure bank, old Bank Number B-0383D

Level: Comprehension

Comments: FHW 12/02 011 2.1.6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Small Reactor Coolant System Leak (cont'd)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**DETERMINE** whether leak stopped because of isolation of charging and letdown.	**RESTORE** CVCS charging and letdown **USING** EA-62-5, Establishing Normal Charging and Letdown.
11.	**EVALUATE** placing excess letdown in service using EA-62-3, Establishing Excess Letdown.	
12.	**INITIATE** leak repairs.	
13.	**GO TO** appropriate plant procedure.	

END OF SECTION

4.0 OPERATOR ACTIONS

4.1 Section Applicability

1. **PERFORM** Section 4.2. ☐

4.2 Placing Excess Letdown in Service

1. **IF** excess letdown is only letdown flowpath,
   THEN
   CONTROL charging flow as necessary to prevent high pressurizer level. ☐

2. **ENSURE** CCS inlet to excess letdown heat exchanger [FCV-70-143] OPEN. ☐

3. **ENSURE** CCS outlet to excess letdown heat exchanger [FCV-70-85] OPEN. ☐

4. **VERIFY** CCS flow to excess letdown heat exchanger greater than 230 gpm, as indicated on [FI-70-84]. ☐

5. **ENSURE** excess letdown divert valve [FCV-62-59] in NORMAL. ☐

6. **OPEN** excess letdown isolation valve [FCV-62-54] . ☐

7. **OPEN** excess letdown isolation valve [FCV-62-55] . ☐

8. **ADJUST** excess letdown flow control valve [FCV-62-56] as necessary to control flow WHILE maintaining heat exchanger outlet temperature less than 200°F, as indicated on [TI-62-58] . ☐

1. OPL271C379.3 001

The plant is operating and the following conditions exists:

Reactor power = 58% slowly increasing
RCS pressure = 2210 psig slowly decreasing
Auctioneered T-avg = 560°F slowly decreasing
Turbine power = 595 MW (steady - no change)
S/G levels = 44% stable
Steam pressure = 900 psig slowly decreasing
Containment pressure = approximately 0.5 psig slowly increasing

Based on the indications listed above, the most likely event in progress is which ONE of the following?

A. LOCA inside containment.

✓B. Steamline break inside containment.

C. Steamline break outside containment.

D. LOCA outside containment.

A. Incorrect, RCS pressure is dropping slowly, reactor power in increasing.
B. Correct with initial conditions.
C. Incorrect, containment pressure is slowly increasing.
D. Incorrect, RCS pressure is dropping slowly, reactor power in increasing, containment pressure is slowly increasing.

K/A[CFR]: 040 AA2.01 [4.2/4.7] [43.5]

Reference: AOP-S.05 symptoms

LP/Objective: OPL271C390 B.2

History: Part B, old Bank Number B-0185

Level: Analysis

Comments: FHW 12/02 040 AA2.01✓

Comment: Reactor power is slowly increasing due to T-Avg slowly decreasing. Containment pressure is above TS limits and slowly increasing. The cause for these symptoms is an accident in containment which drops T-Avg. The correct answer is steamline break inside containment.

3.0 SYMPTOMS AND ENTRY CONDITIONS

3.1 Symptoms

A. Any of the following annunciators may indicate a steamline or feedwater line break or leak:

PANEL XA-55-2C, HEATER DRAINS AND CONDENSATE	
C-7	LS-2-3A CONDENSER HOTWELL LEVEL ABNORMAL
D-7	LS-2-9A CONDENSER HOTWELL LEVEL ABNORMAL
E-7	LS-2-12A CONDENSER HOTWELL LEVEL ABNORMAL

PANEL XA-55-3C, FEEDWATER AND SG LEVELS	
E-2	STM GEN LEVEL ADVERSE SETPOINT

PANEL XA-55-5A, REACTOR COOLANT - STM - FW	
A-6	TS-68-2M/N RC LOOPS T AVG/AUCT T AVG DEVN HIGH-LOW
A-7	FS-3-35A STEAM GEN FEEDWATER FLOW HIGH
B-6	TS-68-2A/B REACTOR COOLANT LOOPS ΔT DEVN HIGH-LOW
C-6	TS-68-2P/Q REAC COOL LOOPS T REF T AUCT HIGH-LOW

PANEL XA-55-5C, VENTILATION	
B-1	TS-30-31 LOWER COMPT TEMP HIGH
B-3	MS-30-241 LOWER COMPT MOISTURE HI

PANEL XA-55-6A, REACTOR PROTECTION AND SAFEGUARDS	
C-2	TS-68-2E OVERTEMP ΔT AUTO TURB RNBK BLK C-3 ROD WTD
D-2	TS-68-2F OVERPOWER ΔT AUTO TURB RNBK BLK C-4 ROD WTD

3.1 Symptoms

PANEL XA-55-6B, REACTOR PROTECTION AND SAFEGUARDS	
A-1	LS-3-39D STM GEN LOOP 1 LOW FW FLOW LOW WATER LEVEL
A-7	FS-3-35B STM GEN LOOP 1 STEAM/FEEDWATER FLOW MISMATCH
B-1	LS-3-52D STM GEN LOOP 2 LOW FW FLOW LOW WATER LEVEL
B-7	FS-3-48B STM GEN LOOP 2 STEAM/FEEDWATER FLOW MISMATCH
C-1	LS-3-94D STM GEN LOOP 3 LOW FW FLOW LOW WATER LEVEL
C-7	FS-3-90B STM GEN LOOP 3 STEAM/FEEDWATER FLOW MISMATCH
D-1	LS-3-107D STM GEN LOOP 4 LOW FW FLOW LOW WATER LEVEL
D-7	FS-3-103B STM GEN LOOP 4 STEAM/FEEDWATER FLOW MISMATCH

B. Deviations or unexpected indications on any of the following may indicate a steam line or feedwater line break or leak:

- Steam flow higher on one or more channels.
- Increase in feedwater flow.
- Deviations on feedwater regulating valves.
- Main feedwater pump speed increasing.
- Increasing reactor power, decreasing T-avg with automatic rod withdrawal.
- Main steam header pressure dropping.
- Increasing containment pressure, temperature, humidity, and sump level.
- Steam generator level dropping.

3.1 Symptoms

C. The following automatic actions may occur due to a steam line or feedwater line break or leak:

- Reactor trip on OT∆T.
- Automatic rod withdrawal due to dropping T-avg.
- Automatic rod withdrawal stop from C-3 and C-4.
- Turbine runback from OT∆T or OP∆T.

3.2 Entry Conditions

None

99. OPL271C367.1 002

Unit 1 is operating at 100% RTP when the operating crew observes the following:

All ice condensers doors open.
RM-90-106 & 112 radiation increasing.
Pzr level at approximately 58% and decreasing slowing.
1A-A charging pump is running.
FCV-62-93 fully open.
RCS pressure 2225 psig and decreasing slowly.
Cntmt pressure at approximately 0.2 psig and increasing slowly.

Which ONE of the following actions should the operating crew perform? Next

A. ~~Immediately decrease charging flow.~~ isolate letdown.

✓B. ~~Immediately~~ start the 1 B-B charging pump.

C. ~~Immediately~~ start a load decrease.

D. ~~Immediately~~ initiate a reactor Trip.

A. Incorrect, decreasing charging flow will not bring pressurizer level back to program in order to determine if charging can keep up with the RCS leak.

B. Correct, starting an additional charging pump will increase charging flow and aid in determining if the pressurizer level can be stabilized.

C. Incorrect, a load decrease will drop the pressurizer level program and make it more difficult to determine if the pressurizer level can be maintained with charging flow.

D. Incorrect, even though a reactor trip is required if the pressurizer level can not be maintained greater than 10%; that determination has yet to be made since no effect has been made to stablize the pressurizer level.

The student may recall the AOP requires controling charging flow; he must understand that an additional charging pump will increase flow as well as taking manual control of FCV-62-93. He must also understand that determining the stability of the pressurizer level is very important and could allow a RCS leak hunt while at power. A reactor trip is not a solution until the stability of the pressurizer level can be determined unable to be maintained greater than 10%.

99. OPL271C367.1 002

K/A[CFR]:	004 2.4.1 [4.3/4.6] [41.10 43.5]
Reference:	AOP-R.05
LP/Objectives:	OPL271C367 B.2
History:	Procedure bank, old Bank Number B-0383C
Level:	Comprehension
Comment:	FHW 12/02 004 2.4.1

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Small Reactor Coolant System Leak

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1.	**CONTROL** charging flow as necessary to maintain pressurizer level greater than 10%.	**IF** pressurizer level is less than 10% or loss of pressurizer level is imminent, **THEN**
		
		PERFORM the following:
		
		IF in MODE 1, 2 or 3, **THEN** **TRIP** the reactor, **INITIATE** Safety Injection, and
		
		GO TO E-0, Reactor Trip or Safety Injection.
		
		➡
		
		IF in Reduced Inventory OR Midloop, **THEN**
		
		GO TO AOP-R.03, RHR System Malfunction.
		
		➡
		
		IF in MODE 4 or 5, **THEN**
		
		GO TO AOP-R.02, Shutdown LOCA.
		
		➡

2.	**MAINTAIN** VCT level greater than 13 % using automatic or manual makeup.	**IF** VCT level can NOT be maintained **THEN** **ENSURE** CCP suction swaps to RWST.

97. OPL271C180.3 001

Which ONE of the following assures the operator that the heat flux hot channel factor upper bound times the normalized ~~axial~~ peaking factor remains within limits in the event of Xenon redistribution following power changes?

A. Maintaining all control rods at the full out position for the core cycle.

B. Maintaining control rods in specified sequence and overlap.

C. Maintaining specified rod insertion limits.

✓D. Maintaining axial flux difference within specified limits.

A. Incorrect per reference. This requirement reduces rod fretting. see TI-28 att 6
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct per reference.
K/A[CFR]: 2.2.25 [2.5/3.7] [43.2]

Reference: TS Bases 3/4.2.1

LP/Objective: OPL271C180 B.3

History: Part B bank, old Bank Number B-0040

Level: Memory

Comments: FHW 12/02 2.2.25

GFES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. R227

$F_{\Delta H}(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. R227

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(X,Y,Z)$ upper bound envelope of F_Q limit specified in the COLR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. R227 R15 R144

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTORS R227

The limits on the heat flux hot channel factor and the nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. The peaking limits are specified in the COLR per Specification 6.9.1.14. R142

96. OPDP-4 002

A main control room annunciator, HIGH PRESS IN AUX BLDG, has sounded on the 1-M-5 panel repeatedly over the past several hours. You eventually determine that it is a nuisance alarm. All compensatory actions have been taken. Which of the following actions should you take to remove this short term nuisance alarm from service?

A. Follow procedure SPP-10.2, "Clearance Procedure", and have the annunciator's input leads lifted and tagged.

B. Follow procedure SPP-~~9.5, "Temporary Alterations~~", issue a TACF to disable the annunciator by lifting the associate d leads at the transmitter.

✓C. Complete Form OPDP-4-1, Disabled Alarm Checklist, including SM signature, and disable the alarm from the MCR.

D. Initiate a maintenance work request to have the annunciator disabled in the communications room.

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

what is communication Room.

K/A {CFR}:	2.4.33 [2.4/2.8] [41.10 43.5]
References:	OPDP-4
LP/Objectives:	OPL271OPSMGMTL B.12
History:	Procedure bank
Level:	Memory
Comments:	FHW 12/02 2.4.33; revised stem to remove similar wording with the correct answer.

1.0 PURPOSE

This procedure establishes the requirements for disabling/enabling alarms, tracking disabled alarms, establishing compensatory monitoring requirements, and the use of disabled alarm identifiers on annunciator windows. This instruction also establishes the requirements for disabling of individual inputs to annunciator windows. Attachments to this instruction provide the required approvals, compensatory monitoring, instructional steps, tracking mechanism, and restoration steps for strict control of this activity.

2.0 SCOPE

This procedure provides administrative instructions to control and identify the status of the control room annunciators. This instruction should not be used to circumvent the permanent design change process. It is intended to apply to alarms or alarm inputs for reasons such as the following:

- to avoid short-term nuisance alarm conditions.
- to obtain a dark board and to restore alarm functions from other inputs when an alarm input is invalid or cannot be promptly cleared.
- to support maintenance or testing activities.

3.0 INSTRUCTIONS

3.1 GENERAL REQUIREMENTS

A. Before an annunciator can be disabled, the action must be reviewed to ensure that it will not result in an unsafe condition for equipment, personnel, or the public. Any Annunciator disablement requiring a 10CFR50.59 review or a Technical Evaluation (as specified in Appendix A) shall have necessary paperwork completed before disabling the alarm.

B. Each alarm input to be disabled will be reviewed to determine its impact on Technical Specifications, TRM, ODCM, FSAR, EOIs, Radiological Monitoring, and Environmental Evaluation equipment.

C. Independent verification is required for removal and replacement of alarm points associated with safety systems.

D. An alarm point may be temporarily placed in service to determine if the condition has cleared or if corrective maintenance was sufficient to correct the deficiency.

E. Numerous alarms receive input from multiple points, any of which may cause the alarm to annunciate. If the alarm does not have "reflash" (i.e., the alarm contacts are "Daisy Chained" in the field), other alarm contacts may be masked while the one alarm contact is in. If possible, the cause of the alarm should be determined, and the individual point removed from scan, or leads disconnected in the field to restore the alarm function from other inputs.

F. Form OPDP-4-1, "Disabled Alarm Checklist" shall be maintained in the Disabled Annunciator Book with a copy of the 50.59 review and Technical Evaluation (if applicable).

G. The Shift Manager/Unit Supervisor shall approve any compensatory monitoring required for annunciators to be disabled.

H. If an annunciator is found to be in a failed state or inoperable, the Unit Supervisor or Shift Manager should refer to the appropriate Technical Specification and/or FSAR sections to evaluate system operability.

3.2 Disabling an Alarm

Employees Disabling Alarms

A. Initiate Form OPDP-4-1 for each alarm to be disabled.

B. Determine if alarm point can be disabled by Operations:

WBN ONLY

If at WBN, determine if alarm point can be disabled using input number by looking up (NODE/Mux/Pt) on any of the following:
- MCR Alarm Printer (Address/Real Address)
- 47W610 series prints
- MCR Plant SSDs

Refer to SOI-55.01 for disabling instructions.

SQN ONLY

If at SQN, determine if alarm point can be disabled using SER number by looking up SER number in 0-SO-55-1.
Refer to 0-SO-55-1 for disabling instructions.

BFN ONLY

If at BFN, software disable functions are NOT available.

C. If alarm cannot be disabled by operations, then contact Site Engineering/System Engineer to determine a method for disabling the alarm point.

D. Complete Form OPDP-4-1, and submit to SM/US for review and approval.

STA/Site Engineering

E. Perform Technical Evaluation (Form OPDP-4-2), if required.

87. GO-2-B.1 003

When restarting the unit after a trip, a NOTE in 0-GO-2 gives specific guidance about when the operations staff is to declare mode 2.

Which ONE (~~1~~) of the following is the correct time to declare MODE 2 per this procedure?

The unit enters mode 2 administratively when:

✓A. the control banks are first withdrawn.

B. the shutdown banks are first withdrawn.

C. the Keff is > .99 and RCS Temperature is > 350°F with Reactor Power less than 5%.

D. reactor power is greater than or equal to 1% and RCS Temperature is > ~~540~~ 350°F.

a. Correct - per NOTE in 0-GO-2. This is an administrative requirement.
b. Incorrect - per NOTE in 0-GO-2. When control banks are first withdrawn
c. Incorrect - per NOTE in 0-GO-2. Mode 2 >= 0.99 <= 5% >= 350°F
d. Incorrect - per NOTE in 0-GO-2. Mode 2 >= 0.99 <= 5% >= 350°F

K/A {CFR}:	2.1.22	[2.8/3.3]	[43.5]
References:	0-GO-2 R14 section 5.2 [11] note.		
LP/Objectives:	OPL271C50 B.1		
History:	Procedure Bank		
Level:	Memory		
Comments:	FHW 12/02 2.1.22		

Unit__________ STARTUP No.__________ Date __________

5.2 **Reactor Startup after a refueling outage** (Continued)

[10] **ENSURE** all shutdown rods are fully withdrawn in accordance with 0-SI-OPS-000-004.0, *Surveillance Requirements Performed on Increased Frequency with no Specific Frequency,* within 15 minutes prior to withdrawing control rods.

__________/________/________
Initials Time Date

[11] **IF** 0-SI-SXX-068-127.0 is not in progress, **THEN**

INITIATE applicable sections of 0-SI-SXX-068-127.0 **PRIOR** to achieving reactor criticality to satisfy SR 4.1.1.4.a., *Minimum Temperature For Criticality.* __________

NOTE The unit enters Mode 2 when the control banks are first withdrawn.

[12] **PERFORM** 0-RT-NUC-000-003.0, *Low Power Physics Testing* to control approach to criticality while continuing with this instruction. ☐

[13] **WHEN** annunciator XA-55-4A, window D-2

P-6 INTERMEDIATE RANGE PERMISSIVE

is **LIT, THEN** **[C.2]**

[a] **RECORD** both source range readings.

N-31 _____CPS N-32 _____CPS __________
Initials

[b] **RECORD** both intermediate range readings.

N-35 _____% RTP N-36 _____% RTP __________
Initials

73. ES-0.2-B.3 001

Unit 1 has had a loss of offsite power and is cooling down using ES-0.2, Natural Circulation Cooldown. RCS temperature is 500 °F. Power has just been restored to the CRDM fans.

Which ONE (1) of the following describes the effect that starting all CRDM fans will have on the cooldown?

A. The fans will aid significantly in removing heat from the upper head region. A GREATER amount of subcooling is procedurally required for cooldown.

✓B. The fans will aid significantly in removing heat from the upper head region. A SMALLER amount of subcooling is procedurally required for cooldown.

C. The fans will **NOT** significantly contribute to the overall upper head cooldown rate. A GREATER amount of subcooling is procedurally required for cooldown.

D. The fans will **NOT** significantly contribute to the overall upper head cooldown rate. A SMALLER amount of subcooling is procedurally required for cooldown.

A. Incorrect, with the CRDM fans inservice more ambient heat is removed from the vessel head which allows for a smaller subcooling margin.

B. Correct, with the CRDM fans inservice more ambient heat is removed from the vessel head which allows for a smaller subcooling margin.

C. Incorrect, the CRDM fans will remove a significant amount of heat as discussed in Westinghouse ERG ES-0.2.

D. Incorrect, the CRDM fans will remove a significant amount of heat as discussed in Westinghouse ERG ES-0.2.

The student must understand the significance of additional cooling on the vessel head and it's contribution to a smaller required subcooling margin.

K/A [CFR]:	E09 EA2.1 [3.1/3.8] [43.5]
References:	EPM-3-ES-0.2
LP/Objectives:	OPL271C382, B.3
History:	Procedure bank
Level:	Comprehension
Comments:	FHW 12/02 E09 EA2.1 ✓

power restored?

EOP Step Number: 14

INITIATE RCS depressurization:

ERG Step Number: 12

Initiate RCS Depressurization:

Purpose:

To initiate depressurization of the RCS while maintaining required subcooling.

ERG Basis:

The pressurizer pressure should periodically be decreased to maintain the reactor coolant and pressurizer pressure-temperature relationship in accordance with the Technical Specifications and the figures shown in the ERG Appendix to this section (described below). The depressurization should be accomplished using pressurizer auxiliary spray or pressurizer PORVs, depending upon whether letdown is in service.

To prevent possible void formation in the upper head, the plant-specific minimum RCS subcooling based on core exit TCs, as described in the Appendix, should be maintained. With the availability of CRDM cooling fans, the total upper head cooldown rate for T-cold plants varies from a maximum of 54°F/hr to about 45°F/hr when the upper head temperature is cooled to 350°F (34°F/hr from the natural circulation cooldown rate of 50°F/hr plus 20°F/hr from the CRDM fans when the upper head temperature is at its highest, 572°F, to 11°F/hr when the upper head temperature is 350°F). For T-hot plants, the total upper head cooldown rate due to both the natural circulation cooldown rate of 25°F/hr (upper head cooldown rate of 10°F/hr) and the availability of CRDM fans (upper head cooldown rate from 21°F/hr at 600°F to 11°F/hr at 350°F) varies from 31°F/hr initially to about 21°F/hr when the upper head temperature is cooled to 350°F. With CRDM fans available, a subcooling margin of at least 50°F should be maintained for both types of plants during depressurization. Without the availability of CRDM fans, T-cold plants must maintain a minimum subcooling of 100°F during depressurization. For T-hot plants without CRDM fans available, the appropriate precautions, as outlined in Appendix, should be taken to ensure that a minimum subcooling of 200°F is maintained at all times during depressurization. However, if problems arise from a more restrictive Technical Specification limit, they will have to be resolved on a plant specific basis (e.g., lower minimum subcooling requirement will result in a longer upper head cool-off time period).

See letter from R. W. Jurgensen to P. S. Check, St. Lucie Cooldown Event Report, OG-57, April 20, 1981, for more detail on the determination of these limits for natural circulation cooldown.

71. ES-0.0-B.5 002

Given the following plant conditions:

- Reactor trip and SI have occurred.
- S/G level in #2 S/G was increasing uncontrollably.
- Crew transitioned to E-3.
- Chemistry and Rad Con report NO activity in ANY S/G.
- Turbine Building AUO reports #2 S/G MDAFW LCV is leaking by.
- The indicator for AFW flow to #2 S/G is determined to be failed low.
- Isolation of #2 S/G MDAFW LCV stops level increase in #2 S/G.
- #2 S/G pressure is stable at 1010 psig.
- E-3 cooldown is in progress.

Which ONE ~~(1)~~ of the following is the correct procedural action to take in response to the above conditions?

A. Continue in E-3 and terminate SI in E-3.

B. Transition directly to ES-1.1 and terminate SI.

C. Transition to E-2. Transition from E-2 to E-1. Transition from E-1 to ES-1.1.

✓D. Transition to ES-0.0. Transition from ES-0.0 to E-1. Transition from E-1 to ES-1.1.

A. Incorrect, E-3 provides steps to mitigate a SGTR.

B. Incorrect, ES-1.1 has entry conditions from E-0, E-1, and FR-H.1.

C. Incorrect, if in E-3 the any S/G pressure dropping in an uncontrolled manner, any S/G pressure less than 140 psig, or any faulted S/G not isolated would be a transition to E-2. These conditions are not listed in the initial conditions.

D. Correct, per EPM-4 ES-0.0 is applicable only if SI is in progress and E-0 has been performed and entered based on Operator judgment.

K/A[CFR]: E01 EA2.1 [3.2/4.0] [43.5]

Reference: ES-0.0, ES-1.1 entry conditions, E-3, EPM-4.

LP/Objective: OPL271C380 B.2

History: Procedure bank

Level: Comprehension

Commets: FHW 12/02 E01 EA2.1

3.11.5 Use of ES-0.0, Rediagnosis

A. ES-0.0, *Rediagnosis*, is unique among the EOPs in that it has no specific transition into it. It is entered strictly based on operator judgment and is applicable only if SI is in progress and E-0 has already been performed.

B. ES-0.0 should be used when the operator has any concern that he may not be in the right EOP based on plant conditions. This is most likely to happen if multiple accidents occur either simultaneously or sequentially.

C. Once entered, ES-0.0 will either transition the operator to ECA-2.1, E-1, E-2, or E-3, or will return him to the procedure and step in effect, depending on diagnostics done within the procedure.

D. If ES-0.0 determines that an operator should be in a certain series of procedures (e.g., E-1 or ECA-1 series), and he is, then he simply returns to the procedure and step in effect.

E. If ES-0.0 determines that an operator should be in a certain series of procedures (e.g., E-3 or ECA-3 series), and he is NOT, then he is sent to either E-1 (if he should be in E-1 or ECA-1 series) or E-3 (if he should be in E-3 or ECA-3 series) to enter the appropriate series at the beginning and work his way through the series normally from that point on.

1.0 PURPOSE

This procedure provides operator actions to terminate safety injection and stabilize plant conditions.

2.0 SYMPTOMS AND ENTRY CONDITIONS

2.1 ENTRY CONDITIONS

E-0	Reactor Trip or Safety Injection: • Specified termination criteria satisfied.
E-1	Loss of Reactor or Secondary Coolant: • Specified termination criteria satisfied.
FR-H.1	Loss of Secondary Heat Sink: • Secondary heat sink reestablished and SI terminated.

3.0 OPERATOR ACTIONS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**DETERMINE** procedure applicability: • **CHECK** any SI pump RUNNING. • **CHECK** CCPIT flow INDICATED. • **CHECK** E-0, Reactor Trip or Safety Injection, previously COMPLETED.	**IF** SI required, **THEN** **PERFORM** the following: a. **ACTUATE** SI. b. **GO TO** E-0, Reactor Trip or Safety Injection. **IF** SI **NOT** required, **THEN** **RETURN TO** procedure and step in effect.
2.	**CHECK** S/G secondary pressure boundary integrity: • Any S/G pressure stable or rising.	**IF** controlled cooldown in progress, **THEN** **GO TO** Step 3. **IF** controlled cooldown **NOT** in progress, **THEN** **PERFORM** one of the following: • **IF** main steamlines **NOT** isolated, **THEN** **GO TO** E-2, Faulted Steam Generator Isolation. OR • **IF** main steamlines isolated, **THEN** **GO TO** ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**CHECK** S/G secondary pressure boundary integrity: • S/G pressures controlled or rising • S/G pressures greater than 140 psig.	**VERIFY** all Faulted S/G(s) ISOLATED: • MSIVs and bypasses CLOSED • AFW ISOLATED • MFW ISOLATED • Atmospheric relief CLOSED • S/G blowdown valves CLOSED • Steam supply to TD AFW pump ISOLATED (S/G 1 or 4). **IF** any Faulted S/G **NOT** isolated, **THEN** **GO TO** E-2, Faulted Steam Generator Isolation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**CHECK** S/G tube integrity: a. **CHECK** the following indications of a S/G tube rupture, including available trends prior to isolation: • Any S/G level rising in an uncontrolled manner. **OR** • Main steamline high radiation. **OR** • Condenser exhaust high radiation. **OR** • S/G blowdown recorder RR-90-120 , pen #1 and pen #2 high radiation. **OR** • Post-Accident Area Radiation Monitor recorder RR-90-268B, points 3 (blue), 4 (violet), 5 (black), or 6 (brown) high radiation. [M-31 (back of M-30)]	**PERFORM** the following: 1) **VERIFY** procedure in effect is one of the following E-1 or ECA-1 series procedures: • E-1, Loss of Reactor or Secondary Coolant. • ES-1.1, SI Termination. • ES-1.2, Post LOCA Cooldown and Depressurization. • ES-1.3, Transfer to RHR Containment Sump. • ES-1.4, Transfer to Hot Leg Recirculation. • ECA-1.1, Loss of RHR Sump Recirculation. • ECA-1.2, LOCA Outside Containment. 2) **IF** E-1 or ECA-1 series procedure in effect, **THEN** **RETURN TO** procedure and step in effect. 3) **IF** E-1 or ECA-1 series procedure **NOT** in effect, **THEN** **GO TO** E-1, Loss of Reactor or Secondary Coolant.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 ENTRY CONDITIONS (Continued)

ECA-3.1 SGTR and LOCA - Subcooled Recovery:
- S/G level rising in an uncontrolled manner.

ECA-3.2 SGTR and LOCA - Saturated Recovery:
- S/G level rising in an uncontrolled manner.

ECA-3.3 SGTR Without Pressurizer Pressure Control:
- S/G level rising in an uncontrolled manner.
- Pressurizer pressure control restored.

FR-H.3 Steam Generator High Level:
- Secondary Radiation.

3.0 OPERATOR ACTIONS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1.0 PURPOSE

This procedure provides actions to terminate leakage of reactor coolant into the secondary system following a steam generator tube rupture.

2.0 SYMPTOMS AND ENTRY CONDITIONS

2.1 ENTRY CONDITIONS

E-0 Reactor Trip or Safety Injection:
- Secondary radiation.
- S/G level rising in an uncontrolled manner.

E-1 Series Foldout Page
- S/G level rising in an uncontrolled manner.

E-1 Loss of Reactor or Secondary Coolant:
- Secondary radiation.
- S/G level rising in an uncontrolled manner.

ES-1.2 Post LOCA Cooldown and Depressurization:
- S/G level rising in an uncontrolled manner.

E-2 Faulted Steam Generator Isolation:
- Secondary radiation.

ES-3.1 Post - SGTR Cooldown Using Backfill:
- S/G level rising in an uncontrolled manner.

ES-3.2 Post - SGTR Cooldown Using Blowdown:
- S/G level rising in an uncontrolled manner.

ES-3.3 Post - SGTR Cooldown Using Steam Dump:
- S/G level rising in an uncontrolled manner.

ECA-2.1 Uncontrolled Depressurization of All Steam Generators:
- Secondary radiation.

(Step continued on next page.)

66. ECA-2.1-B.1 004
ECA-2.1, Uncontrolled depressurization of all Steam Generators is in effect. The following conditions occur:

- TDAFW pump trips on overspeed and can **NOT** be reset.
- "A" MDAFW pump is tagged out.
- "B" MDAFW pump trips on overcurrent.
- Offsite power is lost, with both D/G's re-energizing their respective 6.9 kv AC bus.
- All SG levels are below 0% NR.

The operating team should:

A. Remain in ECA-2.1.

B. Transition to E-0.

C. Transition to E-2.

✓D. Transition to FR-H.1.

A. Incorrect per reference.
B. Incorrect per reference.
C. Incorrect per reference.
D. Correct, the reduction in AFW flow was not by procedure direction therefore a transition to FR-H.1 is required.

K/A[CFR]:	E12 2.4.16 [3.0/4.0] [41.10 43.5]
Reference:	ECA-2.1 note on page 3.
LP/Objectives:	OPL271C419, b.1
History:	Procedure bank, old Bank Number PL-0798
Level:	Analysis
Comments:	FHW 12/02 E12 2.4.16

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION **Isolating both steam supplies to the TD AFW pump when it is the only source of feed flow will result in loss of secondary heat sink.**

NOTE Reducing total feed flow to less than 440 gpm, as directed in this procedure, does NOT require implementation of FR-H.1, Loss of Secondary Heat Sink, as long as a total feed flow <u>capability</u> of 440 gpm is available.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CHECK** secondary pressure boundary:	
	a. **CHECK** the following: • MSIVs and MSIV bypass valves CLOSED • MFW regulating valves and regulating bypass valves CLOSED • MFW isolation valves CLOSED • Atmospheric reliefs CLOSED • S/G blowdown valves CLOSED • MFW flow indication at ZERO.	a. **CLOSE** valves. **IF** valves can **NOT** be closed manually, **THEN** **DISPATCH** personnel to close valves locally, one loop at a time. **IF** MFW flow indicated, **THEN** **CLOSE** additional feedwater or condensate MOVs as necessary.
	b. **CHECK** MD AFW pumps RUNNING.	b. **START** MD AFW pumps. **IF** either MD AFW pump can **NOT** be started, **THEN** **GO TO** Step 2.
	c. **CLOSE** TD AFW pump steam supply valves FCV-1-17 or FCV-1-18.	

65. ECA-1.2-B.2 001

Unit 2 is responding to a LOCA into the Auxiliary Building in ECA-1.2 (LOCA Outside Containment). Upon completion of ECA-1.2, RCS pressure continues to decrease. Which ONE (1) of the following statements correctly describes the correct mitigating strategy to assure continued removal of decay heat under these conditions?

A. Transition back to E-1 (Loss of Reactor or Secondary Coolant).

✓B. Transition to ECA-1.1 (Loss of RHR ~~Containment~~ Sump Recirculation).

C. Transition to ES-1.2 (Post LOCA Cooldown and Depressurization)

D. Transition to ES-1.3 (Transition to RHR Containment Sump).

TRANSFER?

A. Incorrect per reference.
B. Correct per reference.
C. Incorrect per reference.
D. Incorrect per reference.

K/A {CFR}: E04 EA2.1 [3.4/4.3] [43.5]

References: ECA-1.2, p. 7

LP/Objectives: OPL271c418, b.2

History: Procedure bank.

Level: Comprehension

Comments: FHW 12/02 E04 EA2.1.

overlapped with Question 64

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**DETERMINE** if LOCA isolated:	
	a. **CHECK** RCS pressure RISING.	a. **GO TO** ECA-1.1, Loss of RHR Sump Recirculation. ➡
	b. **GO TO** E-1, Loss of Reactor or Secondary Coolant. ➡	

END

61. E15 EA2.1 001

The operating crew is at step two (2) of FR-Z.2, "Containment Flooding."

Which one of the following is the correct criteria necessary to exit FR-Z.2?

A. RHR suction must be aligned to the RWST.

✓B. Containment pressure increased to 3.0 psid.

C. Chem Lab analysis is complete on the RHR system sample.

D. Containment sump was pumped down to < 55%.

A. Incorrect, per reference the only action verb associated with RHR is "check."
B. Correct, per reference. transistion to FR-Z.1.
C. Incorrect, per reference the chem lab is only notified to take a sample.
D. Incorrect, there is no level requirement in the reference.

K/A[CFR]:	E15 EA2.1 [2.7/3.2] [43.5]
Reference:	FR-0 EPM-4
LP/Objective:	OPL271C409 B.1
History:	New question.
Level:	Memory
Comments:	FHW 12/02 E15 EA2.1

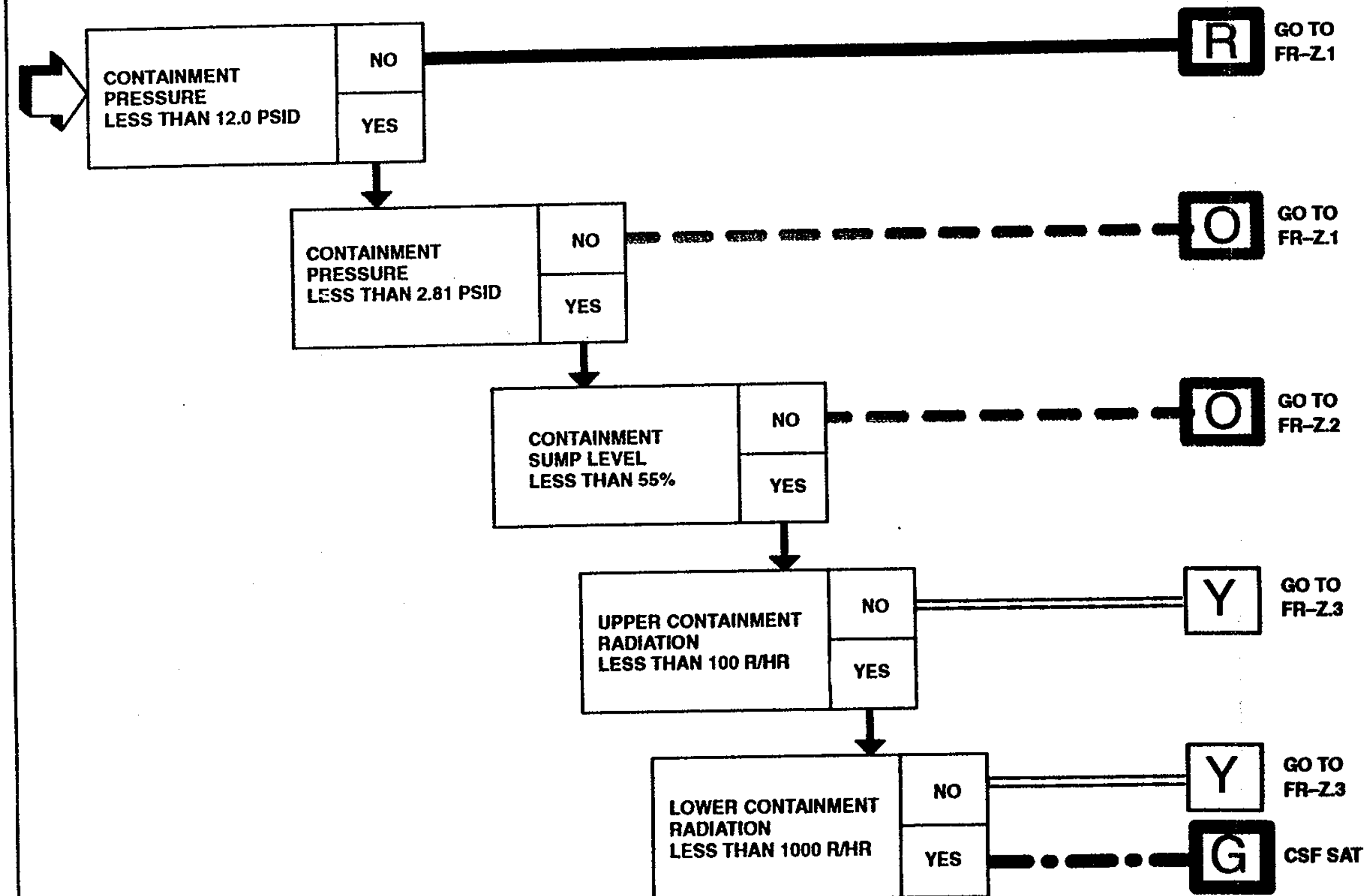
3.10.5 Status Tree Rules of Usage

3. Status trees are designed to monitor for the most severe challenges first to shorten response time in addressing those conditions. Therefore, typically, RED paths will be at the top of a status tree, following by ORANGE, YELLOW, and GREEN as the status tree branches downward.
4. If any RED challenge is detected, the person monitoring status trees informs the procedure reader immediately before continuing with monitoring any subsequent status trees. Since they are monitored in order of importance, the first RED challenge encountered will be the highest priority RED and therefore, the highest priority challenge.
5. If any ORANGE challenge is encountered, the person monitoring status trees continues monitoring until all six status trees have been evaluated. This is necessary because a subsequent RED challenge has priority over any ORANGE challenge. If any RED is encountered, then Rule 3.10.5.D.4 applies. Otherwise, once it is determined that no RED challenges exist, then the person monitoring status trees informs the procedure reader of the highest priority ORANGE challenge.
6. RED or ORANGE challenges must be addressed immediately by implementing appropriate FRPs in order of priority and per the rules of usage. When the person monitoring status trees informs the procedure reader that a RED or ORANGE challenge exists, the procedure reader immediately suspends the ORP (or lower priority FRP) in progress and implements the appropriate FRP, as indicated at the terminus point of the CSF under challenge.
7. YELLOW challenges may be addressed by implementing appropriate FRPs if desired, but do not require immediate operator action. Addressing YELLOW challenges is optional since these are usually temporary, off-normal conditions that will be restored to normal status by actions already in progress. In other cases, the YELLOW path might provide an early indication of a developing RED or ORANGE condition. Following FRP implementation, a YELLOW might indicate a residual off-normal condition. When the person monitoring status trees informs the procedure reader that a YELLOW challenge exists, the procedure reader should evaluate if the YELLOW challenge FRP should be implemented. This decision will be based on the following:

- Whether the procedures in effect will address the challenge as a matter of course.
- Whether the procedures in effect are more important at that time based upon available time and current plant conditions.
- Whether the challenge is of a nature that it will likely develop into an ORANGE or RED condition if action is not taken early.

Containment

F-0.5



63. E16 2.4.41 001

This question has reference material attached.

Plant Conditions: ~~RCS leak present~~
- Chem lab reports RCS activity 280 µCi/gm dose equivalent I-131.
- 1-RM-90-271 and 272 are indicating 370 Rem/Hr.

E-1 has been entered.

Make **NO** assumptions.

Classify this condition per the Emergency Plan Implementing Procedure.

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. General Emergency.

C. correct, EAL 1.1.5.L and 1.3.5.P. Site Area Emergency is a "Loss or Potential Loss of any two barriers"

K/A[CFR]:	E16 2.4.41 [2.3/4.1] [43.5]
Reference:	EPIP-1
LP/Objective:	OPL271C198 B.3
History:	New question.
Level:	Comprehension
Comments:	FHW 12/02 E16 2.4.41 **Provide a copy of EPIP-1**

RM parameters insist RCS leak present.

1.1 Fuel Clad Barrier

1. Critical Safety Function Status

LOSS	Potential LOSS
Core Cooling Red (FR-C.1)	Core Cooling Orange (FR-C.2) **OR** Heat Sink Red (RHR SD cooling not in service) (FR-H.1).

-OR-

2. Primary Coolant Activity Level

LOSS	Potential LOSS
RCS sample activity is greater than 300 µCi/gm dose equivalent Iodine-131	Not Applicable.

-OR-

3. Incore TCs Hi Quad Average

LOSS	Potential LOSS
·eater than 1200 °F on)4-101 OR 102 (EXOSENSOR).	Greater than or equal to 700 °F on XI-94-101 or 102 (EXOSENSOR).

-OR-

4. Reactor Vessel Water Level

LOSS	Potential LOSS
Not Applicable.	VALID RVLIS level < 40% on LI-68-368 or 371 with no RCP running.

-OR-

5. Containment Radiation Monitors

LOSS	Potential LOSS
VALID reading of Greater Than: <u>2.8E + 01</u> Rem/hr On RM-90-271 and 272. **OR** <u>2.9E + 01</u> Rem/hr On RM-90-273 and 274.	Not Applicable.

-OR-

Site Emergency Director Judgment

.ıy condition that, in the judgment of the SM or SED, indicates loss or potential loss of the Fuel Clad Barrier comparable to the conditions listed above.

1.2 RCS Barrier

1. Critical Safety Function Status

LOSS	Potential LOSS
Not Applicable.	Pressurized Thermal Shock Red (FR-P.1). **OR** Heat Sink Red (RHR SD cooling not in service) (FR-H.1).

-OR-

2. RCS Leakage/LOCA

LOSS	Potential LOSS
RCS leak results in subcooling < 40 °F as indicated on XI-94-101 OR 102 (EXOSENSOR).	Non isolatable **RCS** leak exceeding the capacity of <u>one</u> charging pump in the normal charging alignment. **OR** **RCS** Leakage Results in Entry Into E-1.

-OR-

3. Steam Generator Tube Rupture

LOSS	Potential LOSS
SGTR that results in a safety injection actuation. **OR** Entry into E-3.	Not Applicable.

-OR-

4. Reactor Vessel Water Level

LOSS	Potential LOSS
VALID RVLIS level < 40% on LI-68-368 or 371 with no RCP running.	Not Applicable.

-OR-

5. Site Emergency Director Judgment

Any condition that, in the judgment of the SM or SED, indicates loss or potential loss of the **RCS** Barrier comparable to the conditions listed above.

1.3 Containment Barrier

1. Critical Safety Function Status

LOSS	Potential LOSS
Not Applicable.	Containment Red (FR-Z.1) **OR** Actions of FR-C.1 (Red Path) are **INEFFECTIVE** (i.e.: core TC's trending up).

-OR-

2. Containment Pressure/Hydrogen

LOSS	Potential LOSS
Rapid unexplained pressure decrease following initial increase on PdI-30-44 or 45 **OR** Containment pressure or sump level not increasing on LI-63-178 or 179 with a LOCA in progress.	Containment hydrogen increases to > 4% by volume on H_2I-43-200 or 210. **OR** Pressure > 2.81 PSID (Phase B) with no containment spray operating when required (FR-Z.1).

-OR-

3. Containment Isolation Status

LOSS	Potential LOSS
...ntainment isolation, when required, is incomplete and a release path to the environment exists.	Not Applicable.

-OR-

4. Containment Bypass

LOSS	Potential LOSS
Secondary side release outside containment from a **RUPTURED** S/G that cannot be terminated in < 15 minutes (E-2 and E-3). **OR** > 4 hours secondary side release outside containment from a S/G with a S/G tube leak > T/S limits (AOP-R.01, App A).	Unexpected **VALID** increase in area or ventilation RAD monitors adjacent to containment (with LOCA in progress).

-OR-

5. Significant Radioactivity in Containment

LOSS	Potential LOSS
Not Applicable.	**VALID** Reading of greater than: 3.6 E + 02 Rem/hr on RM-90-271 and RM-90-272. **OR** 2.8 E + 02 Rem/hr on RM-90-273 and RM-90-274.

-OR-

6. Site Emergency Director Judgment

Any condition that, in the judgment of the SM or SED, indicates loss or potential loss of the CNTMT Barrier comparable to the conditions listed above.

INSTRUCTIONS

NOTE: A condition is considered to be MET if, in the judgment of the Site Emergency Director, the condition will be MET imminently (i.e., within 2 hours). The classification shall be made as soon as this determination is made.

1. In the matrix to the left, **REVIEW** the Initiating Conditions in all three barrier columns and **CIRCLE** the Conditions that are Met.
2. In each of the three barriers columns, **IDENTIFY** if any Loss or Potential Loss Initiating Conditions have been Met.
3. **COMPARE** the number of barrier Losses and Potential Losses to the Criteria below and make the appropriate declaration.

*NOTE: **MONITOR** the respective status tree criteria if a CSF is listed as an Initiating Condition.*

EMERGENCY CLASS CRITERIA
GENERAL EMERGENCY **LOSS** of any two barriers and **Potential LOSS** of third barrier.
SITE AREA EMERGENCY **LOSS** or **Potential LOSS** of any two barriers.
ALERT Any **LOSS** or **Potential LOSS** of Fuel Clad barrier. OR Any **LOSS** or **Potential LOSS** of RCS barrier.
UNUSUAL EVENT **LOSS** or **Potential LOSS** of Containment Barrier.

Rewrite

59. E-0-B.6 003
Which ONE (1) of the following symptoms would require the initiation of a manual reactor trip **AND** safety injection if neither had occurred automatically?

A. Containment pressure = 1.0 psig

B. General Warning alarm on the Solid State Protection System B

✓C. Pressurizer pressure = 1850 psig, pressurizer level = 40%

D. Power = 33% and loss of flow in one loop

A. Incorrect per reference.
B. Incorrect per reference, general warning requires 2/2 logic.
C. Correct per reference pressurizer pressure < 1970 psig for reactor trip and < 1870 psig for safety injection.
D. Incorrect per E-0, however AOP-R.04 does require a manual reactor trip if greater then P-10 (10%) but it does not require a safety injection.

K/A[CFR]: 013 2.4.47 [3.4/3.7] [41.10 43.5]
2.4.1 (4.3 - 4.6)
2.4.2 (3.9 -4.1)
2.4.4 (4.0 - 4.3)

Reference: E-0

LP/Objective: OPL271C379 B.1

History: Procedure bank

Level: Memory

Comments: FHW 12/02 013 ~~2.4.47~~ 2.4.3

\# TRIVIAL setpoint Question
Not SRO only
K/A mm
Does not use control room reference material.

2.2 SYMPTOMS OF REACTOR TRIP

A. Any valid reactor trip signal [status panel M-5 or M-6].

B. Any reactor trip alarm lit [M-4D].

C. Rapid drop in neutron level indicated by nuclear instrumentation.

D. Shutdown and control rods inserted.

E. Rod bottom lights lit.

F. Rod position indicators at zero.

2.3 SYMPTOMS REQUIRING SAFETY INJECTION

A. Pressurizer pressure less than 1870 psig (blockable below P-11).

B. S/G pressure less than 600 psig (lead/lag) (blockable below P-11).

C. Containment pressure greater than 1.54 psid.

2.4 SYMPTOMS OF SAFETY INJECTION

A. Any valid SI signal [status panel M-6].

B. Any SI alarm lit [M-4D].

C. ECCS pumps running.

3.0 OPERATOR ACTIONS

58. E-0-B.3.A 007

Given the following plant conditions:

- A reactor trip has occurred due to loss of offsite power.
- SI did **NOT** actuate.
- 6.9kV Shutdown Boards 1A-A and 1B-B are energized by the D/Gs.
- The crew has just ensured that all Control Rods are fully inserted.
- The following plant parameters exist:
  - PZR Level is 15% and increasing very slowly.
  - PZR Pressure is 1980 psig and decreasing.
  - AFW Flow is ~580 gpm.
  - All S/G levels are approximately 5%
- The crew ensures that the PZR PORVs are closed and the spray valves are closed.

Which ONE (1) of the following following describes the **MINIMUM** actions that must be performed to stabilize plant conditions?

A. Reset the Blackout Relays on both 6.9kV Shutdown Boards and allow the PZR Pressure Control System to automatically stabilize conditions.

B. Manually actuate SI and return to E-0, "Reactor Trip or Safety Injection".

C. Secure AFW flow to the S/Gs and place the PZR backup heaters control switch to CLOSE.

✓D. Increase PZR level, then allow the PZR Pressure Control System to automatically stabilize conditions.

A. Incorrect, blackout relays can not be reset until offsite power is restored.
B. Incorrect, initial conditions do not require a SI.
C. Incorrect, AFW flow is needed for a heat sink, S/G level >10% per reference.
D. Correct, control charging/letdown so PZR level trending to 25% per reference.

K/A[CFR]: 056 AA2.18 [3.8/4.0] [43.5]

Reference: ES-0.1

Objective: OPL271C381 B.1

History: Procedure bank

Level: Comprehension

Comments: FHW 12/02 056 AA2.18 ✓

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**MONITOR** pressurizer level control:	
	a. **CHECK** pressurizer level greater than 17%.	a. **PERFORM** the following: 1) **ENSURE** letdown ISOLATED. 2) **ENSURE** pressurizer heaters OFF. 3) **CONTROL** charging to restore pressurizer level greater than 17%. 4) **WHEN** pressurizer level greater than 17%, **THEN** **OPERATE** pressurizer heaters as necessary.
	b. **VERIFY** charging IN SERVICE.	b. **ESTABLISH** charging **USING** EA-62-5, Establishing Normal Charging and Letdown.
	c. **VERIFY** letdown IN SERVICE.	c. **WHEN** charging established **AND** pressurizer level greater than 17%, **THEN** **ESTABLISH** letdown **USING** EA-62-5, Establishing Normal Charging and Letdown.
	d. **CHECK** pressurizer level trending to 25% (normal range 20% to 30%).	d. **CONTROL** charging and letdown to maintain pressurizer level at 25% (normal range 20% to 30%).

48. AOP-R.05-B.2 001

Given the following plant conditions:

- Reactor power is STABLE at 90%
- Pressurizer level is DECREASING
- VCT level is DECREASING
- The following annunicators are actuated:
  - PRESSURIZER LEVEL HIGH-LOW
  - LTDN HX OUTLET TO DEMIN TEMP HIGH
  - REGENERATIVE HX LETDOWN LINE TEMP HIGH

Which ONE ~~(1)~~ of the following events would most likely cause these indications?

A. Pressurizer level control valve malfunction.

B. Isolation of CVCS letdown.

✓C. Charging header rupture.

D. Letdown header rupture.

A. Incorrect, both pressurizer and VCT level would not be decreasing.
B. Incorrect, "Regenerative Hx Letdown Line temp High" would not alarm.
C. Correct, with a loss of charging due to a pipe break the pzr would still lower in level due to letdown. Since there is no flow balance VCT level would decrease. Without charging acting as a cooling medium on the regen ht exchgr the temp alarms would annunciate.
D. Incorrect, letdown header rupture would cause a loss of letdown therefore the high temp alarm would not annunciate.

K/A[CFR]: 22 AA2.01 [3.2/3.8] [43.5]

Reference: 1-AR-M6-C(A-4)
1-AR-M6-C(D-4)
1-AR-M5-A(C-3)
AOP-R.05

Objective: OPL271C367, B.2

History: Procedure bank

Level: Analysis

Comments: FHW 12/02 22 AA2.01 ✓

Source	Setpoint	
SER2112 LS-68-335D/E - High SER 2113 LS-68-339F/E - Low	70% span increasing 5 percent of span deviation below level program	**LS-68-335D/E PRESSURIZER LEVEL HIGH-LOW**

Probable Causes

1. RCS leak exceeding charging capacity.
2. Load transient or RCS temperature transient condition.
3. Charging and/or letdown flow mismatch.
4. Instrument malfunction for level or Tavg.

Corrective Actions

[1] **CHECK** pressurizer level (1-LI-68-339A, 335A, 320)

[2] **IF** level is high, **THEN**
ENSURE backup heaters ON.

[3] **ENSURE** level control system is attempting to return level to program with letdown and charging.

[4] **IF** level channel failed, **THEN**
GO TO AOP-I.04, Pressurizer Instrument Malfunction.

[5] **IF** RCS leakage is suspected, **THEN**
GO TO AOP-R.05, RCS Leak and Leak Source Identification.

[6] **IF** in MODE 4 or MODE 5 and a LOCA is identified, **THEN**
GO TO AOP-R.02, Shutdown LOCA (MODE 4, or 5).

[7] **EVALUATE** Technical Specifications 3.3.1, 3.3.2 and 3.4.6.2 as applicable.

References

45B655-05A-0,
47B601-68-45

Source	Setpoint	TS-62-71 REGENERATE HX LETDOWN LINE TEMP HIGH
SER 2172 1-TS-62-71	397°F increasing	

Probable Causes

1. Letdown flow is too high.
2. Charging flow is too low.

Corrective Actions

[1] CHECK letdown temperature on [1-TI-62-71].

[2] CONSIDER the following operations to reduce letdown temperature:

[a] REDUCE letdown flow.

[b] MAINTAIN letdown pressure at ~340 psig.

[c] INCREASE charging flow.

[3] MONITOR reactor coolant pump seal flow.

[4] MONITOR pressurizer level by observing [1-LI-68-335A].

References

45B655–06C–0,
47B601-62-13,
47W610-62-2,
47W809-1

Source	Setpoint	
SER 2160 1-TS-62-78	132°F	**TS-62-78 LTDN HX OUTLET TO DEMIN TEMP HIGH**

Probable Causes

1. Letdown flow is greater than charging flow.
2. Inadequate component cooling H_2O flow through heat exchanger.
3. Failure of 1-TCV-70-192 to respond in AUTO.

Corrective Actions

[1] **CHECK** letdown temperature on [1-TI-62-78].

[2] **ENSURE** letdown demins bypassed at ≥ 137°F to prevent resin damage.

[3] **ENSURE** proper valve alignment in accordance with 1-SO-62-1, *Chemical and Volume Control System.*

NOTE

Automatic positioning of 1-TCV-70-192 is controlled by input from HIC-62-78 (in Auto) and TS-62-78 (In Auto or Manual at >124°F).

[4] **IF** [1-HIC-62-78] not operating properly in **AUTO, THEN**
PLACE in **MANUAL, AND**
ATTEMPT temperature control.

[5] **VERIFY** low pressure letdown valve maintaining letdown pressure at ~ 340 psig.

[6] **IF** Letdown heat exchanger temperature cannot be controlled, **THEN**
EVALUATE switching to excess letdown in accordance with 1-SO-62-6, Excess Letdown.

References

45B655–06C–0,
47W610-62-2

3.0 SYMPTOMS AND ENTRY CONDITIONS

3.1 Symptoms

A. Any of the following annunciators may indicate a RCS leak:

PANEL XA-55-4B, NIS/ROD CONTROL	
E-7	TS-68-398 REACTOR VESSEL VENT TEMP HI

PANEL XA-55-5A, REACTOR COOLANT - STM - FW	
A-1	TS-68-21 REACTOR VESSEL FLANGE LEAKOFF TEMP HIGH
B-1	LS-68-300A/B PRESSURIZER RELIEF TANK LEVEL HI-LOW
B-5	LS-77-125E CNTMT FLOOR & EQUIP DRAIN SUMP HI-HI-HI
C-1	TS-68-309 PRESSURIZER RELIEF TANK TEMP HIGH
C-3	LS-68-335D/E PRESSURIZER LEVEL HIGH-LOW
C-5	LS-77-410A REACTOR BLDG AUX FL & EQ DRAIN SUMP HI
D-2	TS-68-328 PRESSURIZER SAFETY VALVE LINES TEMP HIGH
D-4	PS-68-340G/F PRESSURIZER PRESSURE LOW BACKUP HTRS ON
E-2	TS-68-331 PRESSURIZER POWER RELIEF LINE TEMP HIGH
E-3	LS-68-335E/D PRZR LVL LOW HEATER OFF & LETDOWN SECURED

PANEL XA-55-5C, VENTILATION	
B-1	TS-30-31 LOWER COMPT TEMP HIGH
B-3	TS-30-241 LOWER COMPT MOISTURE HI
B-4	TS-30-240 LOWER COMPT MOISTURE HI
B-6	XS-68-363 PRESSURIZER RELIEF VALVE OPEN

3.1 **Symptoms (cont'd)**

PANEL XA-55-6C, CVCS-HEAT TRACE-UHI	
A-3	LS-62-129A/B VOLUME CONTROL TANK LEVEL HI-LOW
C-5	CONTAINMENT VENTILATON ISOLATION TRAIN A
C-6	CONTAINMENT VENTILATON ISOLATION TRAIN B
E-3	LS-63-104 CONTAINMENT SUMP FULL
E-4	LS-63-176 CNTMT LEVEL HI RHR RECIRC
E-7	FCV-74-1/2 TROUBLE OR RHR PRESS HI

PANEL 1(2)-XA-55-6D, AUXILIARY SYSTEMS	
A-4	(2-)PS-74-13 RHR PUMPS DISCH PRESS HI OR MINIFLOW CONDITION
E-1	AUX BLDG HIGH ENERGY LINE BREAK

PANEL XA-55-30, POST ACCIDENT RADIATION MONITORING	
A-1	RA-271A UPPR IN CNTMT HI RAD
A-2	RA-273A LWR IN CNTMT HI RAD
B-1	RA-272A UPPR IN CNTMT HI RAD
B-2	RA-274A LWR IN CNTMT HI RAD

PANEL 0-XA-55-12A, UNIT 1 AND COMMON RADIATION MONITOR	
A-4	1-RA-90-106A CNTMT BLDG LWR COMPT AIR MON HIGH RAD
B-1	1-RA-123A CCS LIQ EFF MON HIGH RAD

PANEL 0-XA-55-12B, COMMON RADIATION MONITOR	
A-7	1-RA-90-59A RX BLDG AREA RAD MON HIGH RAD
C-5	0-RA-123A CCS LIQ EFF MON HIGH RAD

PANEL 0-XA-55-12D, UNIT 2 AND PLNT LIQ DISCH RADMON	
A-4	2-RA-90-106A CNTMT BLDG LWR COMPT AIR MON HIGH RAD
B-1	2-RA-123A CCS LIQ EFF MON HIGH RAD
B-3	2-RA-90-59A RX BLDG AREA RAD MON HIGH RAD

3.1 Symptoms (cont'd)

B. Deviations or unexpected indication on any of the following may indicate an RCS leak:

1. Rising charging flow to maintain pressurizer level.
2. Rising makeup flow to VCT.
3. Rising containment pressure.
4. Dropping pressurizer level.
5. Rising pressurizer level (leak in vapor space).
6. Rising containment moisture.
7. Rising containment radiation.
8. Rising HELB recorder indications.

C. Any of the following automatic actions may indicate an RCS leak:

1. CCS high radiation will cause the surge tank vent FCV-70-66 to close.
2. RCS leak in auxiliary building may cause auxiliary building isolation and ABGTS to start.
3. RCP thermal barrier cooling coil leak will cause CCS flow to isolate on high differential flow.
4. Ice condenser door(s) opening.

3.2 Entry Conditions

None

END OF SECTION

47. AOP-R.04-B.5 001
Unit Two is operating at 30 % RTP when Loop 3 RCP trips.

Which ONE ~~(X)~~ of the following describes the initial unit response?

A. ~~A reactor trip will occur and~~ There will be no flow in loop 3.

CORE ΔP WILL DECREASE CAUSING OPERATING LOOPS FLOW TO INCREASE

B. ~~A reactor trip will occur and~~ loop 3 flow will decrease. ~~AND STABILIZE TO ABOUT~~ APPROX 20
CORE ΔP WILL INCREASE CAUSING OPERATING LOOPS FLOW TO INCREASE

C. ~~A reactor trip will NOT occur and~~ there will be no flow in loop 3.
CORE ΔP WILL INCREASE SLIGHTLY CAUSING OPERATING LOOPS FLOW TO INCREASE SLIGHTLY

✓D. ~~A reactor trip will NOT occur and~~ loop 3 flow will decrease TO APP 20%
CORE ΔP WILL DECREASE SLIGHTLY CAUSING OPERATING LOOPS FLOW TO INCREASE SLIGHTLY

A. Incorrect, the reactor will not trip as initial unit response, the unit is below P-8 logic. AOP-R.04 will require a manual reactor trip.
See "D"

B. Incorrect, the reactor will not trip as initial unit response, the unit is below P-8 logic. AOP-R.04 will require a manual reactor trip.
SEE "D"

C. Incorrect, the reactor is below P-8 auto trip logic and the initial unit response is NOT a reactor trip. Affected loop flow decreases to zero and then reverses to 20-25% of nominal flow caused by the delta-P across the reactor vessel. Therefore no flow is incorrect.
SEE "D"

D. Correct, affected loop flow decreases and then reverses to 20-25% of nominal flow caused by the delta-P across the reactor vessel.

K/A[CFR]:	015 AA2.07 [2.1/2.9] [43.5]
Reference:	AOP-R.04, RCP system description, abnormal operations.
LP/Objective:	OPL271RCP B.13
History:	Procedure bank
Level:	Analysis
Comments:	FHW 12/02 015 AA2.07

47. Unit Two is operating at 30% RTP when Loop 3RCP trips. Which ONE of the following describes the initial unit response?

A. There will be no flow in loop 3; Core DP will **decrease** causing operating loops flow to **increase.**

B. Loop 3 flow will **decrease** to approximately 20%; Core DP will **increase** slightly causing operating loops flow to **decrease** slightly.

C. There will be **negative** flow in loop 3; Core DP will **increase** slightly causing operating loops flow to **increase** slightly.

D. Loop 3 flow **decrease** to approximately 20%; Core DP will **decrease** slightly causing operating loops flow to **increase** slightly.

Author: rfa

Abnormal Operations

Loss of a RCP below P-8

Upon a loss of a single RCP, plant parameters will change as described in the table below. The loss of two or more RCPs above P-7 and below P-8 results in a reactor trip.

Paremeter	Plant Response
Turbine Power	Reduces slightly as P_{stm} lowers.
Reactor Power	Reduces slightly as turbine power reduces.
Core ΔT	Increases due to decreased mass flow through the reactor.
T_{avg} (affected loop)	Decreases to T_C due to backflow in the loop caused by the delta-P across the reactor vessel.
T_{avg} (unaffected loops)	Remains the same.
Steam Generator Pressure (affected loop)	Decreases due to decrease in affected loop T_{avg} and hence T_{STM} .
Steam Generator Pressure (unaffected loops)	Decreases since each unaffected loop has to produce more power. Therefore, (T_{avg} - T_{STM}) increases. Since T_{avg} remains the same; T_{STM} must decrease. If T_{STM} decreases; P_{STM} decreases.
ΔT (affected loop)	Decreases due to reduction in steam generation in the loop.
ΔT (unaffected loops)	Increases due to each unaffected loop producing significantly more power.
Loop Flow (affected loop)	Affected loop flow decreases to zero and then reverses to 20-25% of nominal flow caused by the delta-P across the reactor vessel.
SG Feedwater Flow (affected loop)	Decreases as steam flow from the affected SG decreases.
SG Feedwater Flow (unaffected loops)	Increases as steam flow from the un affected SGs increase.

Continued on next page

0K

1. AOP-M.03-B.1 002

   Given the following plant conditions:

   - Unit Two in MODE 3 for maintenance.
   - Panel 0-XA-55-27B-D Annunciator A-4,"MISC EQUIP SUP HDR FLOW LOW", starts alarming.
   - Panel 0-XA-55-27B-D Annunciator A-6, "LETDOWN HX OUTLET FLOW/TEMP ABNORMAL", starts alarming.

   Which ~~ONE (1)~~ of the following events could cause both alarms to actuate?

   ✓A. CCS supply header rupture.

   B. Letdown HX tube rupture.

   C. Loss of seal injection.

   D. Loss of charging flow.

   A. Correct because CCS supply header rupture causes a loss of cooling to the letdown heat exchanger.

   B. Incorrect because the combination of annunciators A-4 and A-6 indicate a CCS failure instead of a letdown heat exchanger failure.

   C. Incorrect because the combination of annunciators A-4 and A-6 indicate a CCS failure instead of a loss of seal injection.

   D. Incorrect because the combination of annunciators A-4 and A-6 indicate a CCS failure instead of a loss of charging flow.

K/A {CFR}:	APE 026 AA2.01	[2.9/3.5]	[43.5]	Both are ok
	2.4.45	[3.3/3.6]	[43.5]	

References: AOP-M.03 "Loss of Component Cooling Water'
0-AR-M27-B-D (A-4)
0-AR-M27-B-D (A-6)

LP/Objectives: OPL271C425, B.1

History: 98 NRC Exam

Level: Comprehension not exactly

Comments: FHW 12/02 2.4.45

3.0 SYMPTOMS AND ENTRY CONDITIONS

3.1 Symptoms

A. Any of the following annunciators may indicate a CCS failure:

PANEL 0-XA-55-12A, UNIT 1 AND COMMON RADIATION MONITOR	
B-1	1-RA-90-123A CCS LIQ EFF MON HIGH RAD

PANEL 0-XA-55-12D, UNIT 2 AND PLT LIQ DISCH RADMON	
B-1	2-RA-90-123A CCS LIQ EFF MON HIGH RAD

Source	Setpoint	
SR 1204 2-FS-70-164A	300 gpm decreasing	MISC EQUIP SUP HDR FLOW LOW
Retransmitted to U-2 SER 2179		

Probable Causes

1. Component cooling pump tripped.
2. Inadvertent MOV closure.
3. Manual valve misaligned.
4. Pipe Break.

Corrective Actions

[1] CHECK miscellaneous equipment supply header flow by observing [2-FI-70-164A].

[2] CHECK indicating lights to verify CCS pump running.

[3] VERIFY proper valve alignment in accordance with 2-SO-70-1, *Component Cooling Water System Train A.*

[4] IF component cooling water loss is indicated, THEN GO TO AOP-M.03, *Loss of Component Cooling Water.*

NOTE The 50.59 safety assessment requirement for disablement of this annunciator is not applicable during unit outage. (SA/SE requirement of 0-PI-OPS-055-001.0 Appendix A.

[5] IF component cooling water low flow due to Unit outage, THEN
DISABLE this annunciator in accordance with 0-PI-OPS-055-001.0 Appendix A.

References 45B655-27BD-0, 47B601-70-43, 47W610-70-2, 47W859-1

LETTDOWN HX OUTLET FLOW/TEMP ABNORMAL

Source	Setpoint
SER 1230	
2-FS-70-190	100 gpm decreasing
SER 1231	132°F increasing
2-TS-70-191	
Retransmitted to U-2	
SER 2201 and 2202	

Probable Causes

1. CCS pump off.
2. Heat exchanger isolation valves closed or misaligned.
3. Low CCS water pressure.
4. 2-TCV-70-192 failed closed.
5. Pipe break.

Corrective Actions

[1] CHECK letdown heat exchanger outlet flow by observing [2-FI-70-190] and outlet temperature by observing [2-TI-70-191].
[2] MONITOR CVCS letdown temperature on [2-TI-62-78].
[3] IF [2-TCV-70-192] not operating properly in AUTO, THEN
PLACE in MANUAL, AND
ATTEMPT temperature control.
[4] VERIFY low pressure letdown valve maintaining letdown pressure at ~ 340 psig.
[5] IF Letdown heat exchanger temperature cannot be controlled, THEN
EVALUATE switching to excess letdown in accordance with 2-SO-62-6, *Excess Letdown.*
[6] VERIFY proper valve alignment in accordance with 2-SO-70-1, *Component Cooling Water System Train A.*

NOTE 2-TCV-70-192 fails open on loss of air or high-high temperature of 124°F on 2-TIS 62-79, but could fail closed on a circuit failure of loop 2-T-62-78.

[7] IF this annunciator clears and re-alarms, AND [2-TCV-62-79] has diverted to the VCT, THEN
NOTIFY Instrument Maintenance to troubleshoot instrument loop 2-T-62-78.
[8] IF pipe break is suspected, THEN
GO TO AOP-M.03, *Loss of Component Cooling Water.*
[9] START additional CCS pumps in accordance with 2-SO-70-1, as necessary.
[10] IF maintenance is required, THEN
NOTIFY Work Control Group for support.
[11] IF alarm in due to Letdown HX being out of service, THEN
Continued operation with alarm in is acceptable.

References 45B655-27BD-0, 47B601-70-49, 47W610-70-2, 47W859-3.

40. AOP-I.04-B.2 001

During Mode 1 operation, with the Pressure Control Selector Switch selected to the NORMAL position (340/334), which ONE ~~(1)~~ of the following statements describes the effects of Pressurizer Pressure Control Channel I failing LOW? Assume no operator action is taken.

✓A. All heaters on; LOW PRESSURE alarm; PORV-340A blocked; RCS pressure rises; PORV-334 opens at 2335 psig.

B. Sprays full on; PORV-340A opens; Low Pressure Reactor Trip at 1970 psig; Low Pressure SI at 1870 psig.

C. All heaters on; LOW PRESSURE alarm; PORV-340A setup for open signal; RCS pressure increases to 2335 psig.

D. Sprays full on; PORV-340A blocked; all heaters on at 2210 psig; heaters modulate to maintain RCS pressure between 2210 and 2218 psig.

A. Correct per reference.
B. Incorrect, PZR press channel failing low will not fully open the sprays.
C. Incorrect, channel I failing low does not set up PORV 340A for opening.
D. Incorrect, PZR press channel failing low will fully open the sprays.

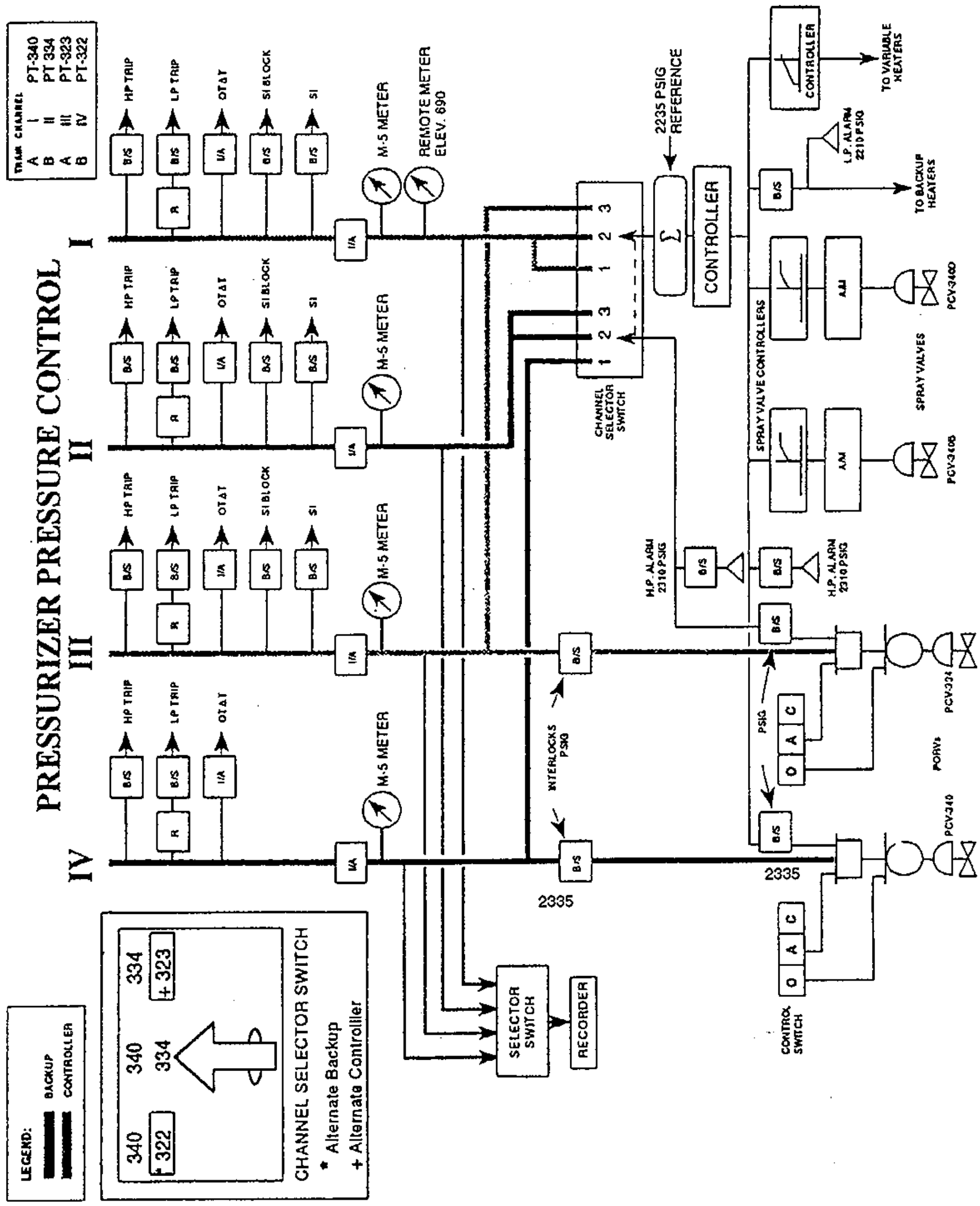
ok K/A[CFR]: 010 2.4.47 [3.4/3.7] [41.10 43.5]
ok 027 AK2.03 (2.6 - 2.8)
~~027 AK2.03 (2.6/2.8)~~
ok 027 AA1.01 (4.0 - 3.9)
No 027 AA2.16 (3.6 - 3.9)
~~027 AA1.01 (4.0/3.9)~~
~~027 AA2.16 (3.6/3.9)~~

Reference:	AOP-I.04
LP/Objective:	OPL271PZRPCS B.11
History:	Procedure bank. old Bank Number PL-0914
Level:	Analysis
Comments:	FHW 12/02 010 2.4.47.

Had one question for low high?

APPENDIX H

PRESSURIZER PRESSURE CONTROL



39. AOP-C.04-B.5 008

Due to conditions causing Control Room Inaccessibility, the main control room has been abandoned and all checklists are complete. Hot Standby conditions are being maintained from the Auxiliary Control Room when 2B-B 6.9-kV S/D Bd. experiences a loss of voltage.

Which ONE (X) of the following is the expected response by the operating staff for this condition?

A. Check Diesel Generators running and all auto-connected to the 6.9-kV S/D Boards.

✓B. Ensure Diesel Generators running and dispatch personnel to manually close the OK 2B-B 6.9-kV S/D Board Emergency Feeder Bkr. Verify 2B-B D/G connected to 2B-B 6.9-kV S/D Bd. OK

C. Verify Diesel Generators running and dispatch personnel to manually close all 6.9-kV S/D Bd. Emergency Feeder Breakers. Verify all D/Gs connected to the 6.9-kV S/D Boards.

D. Verify D/Gs running and 2B-B D/G auto-connected to the 2B-B 6.9-kV S/D Board.

A. Incorrect - Diesels will not auto connect after checklist have been completed. Switches are in auxiliary position.
B. Correct - Diesels will not auto connect after checklist have been completed. Switches are in auxiliary position and breaker must be closed locally.
C. Incorrect - Only ONE SD BD has lost voltage.
D. Incorrect - Diesels will not auto connect after checklist have been completed. Switches are in auxiliary position.

K/A {CFR}:	2.3.3	[1.8/2.9]	{43.4}
References:	AOP-C.04 section 2.1 [13] note		
LP/Objectives:	OPL271C423 B.5		
History:	Procedure bank		
Level:	Memory		
Comments:	FHW 12/02 2.3.3; since the SRO is the procedure reader it is his responsibility to ensure actions are completed.		

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.1 Control Room Abandonment (cont'd)

NOTE Operations Duty Specialist phone number is 751-1700.

12. **NOTIFY** Shift Manager to activate REP **USING** EPIP-1, Emergency Plan Classification Matrix.

NOTE 1 D/G breakers will not automatically close when transfer switches are in AUXILIARY. If off-site power is lost, D/Gs should automatically start but D/G breakers must be manually closed to energize shutdown boards.

NOTE 2 Breaker position indication lights inside ACR are not available until associated transfer switches are placed in AUX position.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**MONITOR** shutdown boards ENERGIZED from start busses. [0-L-4]	a. **ENSURE** affected shutdown board(s) energized from D/G as follows: 1) **ENSURE** D/Gs running. [0-L-4] 2) **NOTIFY** operator in 6.9KV Shutdown Board Rm to ensure affected shutdown board(s) energized from D/G **USING** Checklist 3 (Unit 1) or Checklist 4 (Unit 2). 3) **NOTIFY** operator at D/G Bldg to ensure D/G ERCW valves OPEN. b. **WHEN** personnel available, **THEN** **REFER TO** AOP-P.01, Loss of Offsite Power.

37. AOP-C.01-B.5 011

Given the following:

- Unit 1 was operating at 83% power.
- A control rod dropped into the core
- One negative rate trip status light came on
- One minute later, a second rod dropped into the core
- NO additional rate trip status lights are lit

Which ONE of the following actions is required?

A. Recover the dropped rods using AOP-C.01, Rod Control System Malfunctions.

B. Shutdown the reactor using AOP-C.03, Emergency Shutdown.

✓C. Trip the reactor and go to E-0, Reactor Trip or Safety Injection.

D. Reduce power and enter AOP-I.01, Nuclear Instrument Malfunction.

This Q has NO discriminatory value

A. Incorrect per reference.
B. Incorrect per reference.
C. Correct per reference.
D. Incorrect per reference.

K/A: 2.4.11 (3.4/3.6 {41.10, 43.5}
~~2~~.4.4 (4.0/4.3) {41.10, 43.2}
003 AK3.04 (3.8/4.1) {41.5, 41.10}
003 AA2.03 (3.6/3.8) {43.5}
001 K1.05 (4.5/4.4) {41.2-9}

Reference: AOP-C.01, Section 2.3, page 13

Objective: OPL271AOPC01, b.5

Level: Memory

History: Procedure bank

Comments: FHW 12/02 003 ~~2~~.4.4.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2.3 Dropped Rod(s)

1.	**DETERMINE** number of dropped rods LESS THAN TWO.	**TRIP** the reactor and **GO TO** E-0, Reactor Trip or Safety Injection. [C.5] →
2.	**PLACE** rod control in MAN.	

NOTE: Reducing turbine load to match T-Ref to T-avg. is the preferred action. Rod withdrawal is allowed.

3. **ESTABLISH** T-avg. at T-Ref by either of the following methods:

   - **REDUCE** turbine load

     OR

   - **WITHDRAW** control rods

NOTE: If the reactor is subcritical, retrieval of the dropped rod is NOT the conservative action to take. Opening the reactor trip breakers is preferred.

4.	**CHECK** reactor CRITICAL. **[C.2] [C.5]**	**TRIP** the reactor and **GO TO** E-0, Reactor Trip or Safety Injection. →

24. 069 2.1.14 001

This question has reference material attached.

Unit 1 is at 100% RTP. 0-SI-SLT-000-160.0, "Primary Containment Total Leak Rate," has just been completed for Unit 1. System Engineering reports to the Unit 1 US that the combined bypass leakage rate is 65 scfh. L_a is 225.17 scfh.

Which one of the following should be notified of the surveillance results?

A. No additional notifications are required

✓B. Plant Manager, Senior Vice President Nuclear Operations, Duty Plant Manager

C. Site Vice President Nuclear Operations, Duty Plant Manager, Operations Duty Specialist

D. Plant Manager, Site Vice President Nuclear Operations, Operations Duty Specialist.

A. Incorrect. Analysis of the data (65/225.17 > .25) reveals that Unit 1 should enter LCO 3.6.1.2. This is a four hour LCO. Therefore, SPP-3.5 Appendix C requires site notifications for an unplanned entry into a LCO with a time duration of 72 hours or less.

B. Correct. Condition describes entry into a LCO with duration of less than 72 hours. Therefore, the Plant Manager, Senior Vice President Nuclear Operations since LCO less than 24 hours, and the Duty Plant Manager are required to be notified. A TS safety limit was not exceeded; therefore, the ODS is not required to be notified. Per the * note, the Plant Manager notifies the Site Vice President.

C. Incorrect. Justification per "B" above.

D. Incorrect. Justification per "B" above.

K/A[CFR]: 069 2.1.14 [2.5/3.3] [43.5]

References: TS 3.6.1.2 and SPP-3.5 Rev.10, Appendix C

LP/Objective: OPL271C168 B.3

History: New Question

Level: Comprehension.

Comments: FHW 12/02 069 2.1.14; Written by Jim Kearney (9/11/02).
Provide TS 3.6.1.2 and SPP-3.5 Rev 10, Appendix C.

SECONDARY CONTAINMENT BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Secondary Containment bypass leakage rates shall be limited to a combined bypass leakage rate of less than or equal to 0.25 percent L_a for all penetrations that are secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.* R207 R221

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: R180

With the combined bypass leakage rate exceeding 0.25 L_a for BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, restore the combined bypass leakage rate from BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING to less than or equal to 0.25 L_a within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when Secondary Containment Bypass Leakage results in exceeding the overall containment leakage rate acceptance criteria. R221

 R180

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated: R180

a. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests in accordance with the Containment Leakage Rate Test program, except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test. R221 R180

b. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

c. The provisions of Specification 4.0.2 are not applicable.

R221

APPENDIX C
Page 1 of 1
SITE EVENT NOTIFICATION MATRIX

	Notification Requirements			
Event/Condition	Plant Manager	SVP, Nuclear Operations	Duty Plant Manager	Ops. Duty Spec. (ODS)
Reactor/Turbogenerator trip, unscheduled unit power reduction, or nonscheduled unit shutdown; and when unit is restored to full service.	Yes*	Yes	Yes*	Yes
Unplanned entry into a Limiting Condition for Operation with time duration of 72 hours or less.	Yes*	Only for duration of 24 hours or less	Yes*	Only when TS safety limits exceeded
Classification of the Radiological Emergency Plan (REP) at any entry level.	Yes*	Yes	Yes*	Yes
Personnel injuries that are potential lost-time injuries, serious recordable injuries, and injuries requiring hospital admittance or transport to an offsite medical facility.	Yes*	Yes	Yes*	Yes
Death of any person as a result of injuries received on site or due to medical problems occurring while onsite.	Yes*	Yes	Yes*	Yes
Release of oil or hazardous materials to the discharge canal, ponds or river and violations of the NPDES permit.	No	No	Yes	Yes
Any event which may be newsworthy to the public. (1)	Yes*	Yes	Yes*	Yes
NRC 1 hour, 4 hour, or 8 hour phone calls.	Yes*	Yes	Yes*	Yes
Any unusual radiation exposure to personnel.	Yes*	No	Yes*	Yes
Accidental, unplanned or uncontrolled radioactive release.	Yes*	No	Yes*	Yes
Any reasonable threat to generation.	Yes	Yes	Yes	No
Outage critical path extensions exceeding 12 hours.	Yes	Yes	Yes	No
Any reactivity event or unplanned reactivity change.	Yes	Yes	Yes	No

NOTE:
(1) Consider items specified in Appendix D step 2.2 of this procedure.
* Plant Manager should ensure the Site Vice President (VP) is notified. If Site VP can not be contacted within 30 minutes, the SVP, Nuclear Operations should be notified. If the SVP, Nuclear Operations can not be reached, the Chief Nuclear Officer should be notified.