Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 1	Group #	1	
	K/A #	000007EA1.04	
OK	Importance Rating	3.6	

Ability to operate and monitor RCP operation and flow rates as they apply to a reactor trip

Proposed Question: Common 1

Given the following conditions:

- The plant is operating at 25% power
- The high pressure tap to RCS flow instrument FT-416A on loop 31 develops a large leak

What is the resulting plant condition, if NO operator action is taken?

A. All loop 31 flow indicators will read low, but the reactor trip is NOT generated on RCS loop low flow.

B. All loop 31 flow indicators will read low, but the reactor trip is generated on low PRZR pressure.

C. Only FI-416A RCS flow indication will read low, but the reactor trip is NOT generated on RCS loop low flow.

D. Only FI-416A RCS flow indication will read low, but the reactor trip is generated on low PRZR pressure.

Proposed Answer:

A. All loop 31 flow indicators will read low, but the reactor trip is NOT generated on RCS loop low flow.

Explanation (Optional):

Technical Reference(s): SD-1.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCS-001 0002c (As available)

Question Source: Bank # INPO
Modified Bank # 19403 (Note changes or attach parent)
New

Question History: 12/11/2000 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2,3,7,14
55.43

Comments:

WS #1

443 connected to the loop 34 hot leg via a common root line isolated by manual valve RC-512.

PI-475 is a locally indicating, drag type pressure indicator with a range of 0-2500 psig. It is mounted adjacent to steam generator 34 on the 46-foot elevation in containment.

PT-403 is a wide range transmitter that supplies a pressure signal to:

- the plant computer,
- recorder PR-402/3 (range 0-3000 psig) on panel FDF,
- indicator PI-403 (range 0-3000 psig) on panel FDF,
- the RCS saturation monitor,
- RHR suction valve RH-MOV-731 motor operator.
  - The signal to RHR suction valve RH-MOV-731 is an interlock signal. MOV-731 cannot be opened until PT-403 is less than 450 psig and the valve auto closes if PT-403 exceeds 550 psig. (RHR Suction Valves, RH-MOV-730 and RH-MOV-731 are located in series so that closure of either valve isolates the RHR system from the RCS.)

PT-443 provides pressure input signals to the OPS, the plant computer, and to PI-443K (0-1500psig range) on panel SFF.

LOOP 33

The loop 33 pressure transmitters are shown on Figure 1.1-25 and include PT-433 and PI-1400.

PT-433, which taps off the loop 33 hot leg sample line via manual isolation valve SP-1026. PT-433 supplies PI-433K (0-1500 psig range) on panel SFF, and provides pressure input signals to the OPS and to the plant computer.

Figure 1.1-25 also shows PI-1400, a 0-3000 psig locally indicating pressure gauge. PI-1400 is installed on the combined Loop 31 and 33 hot leg sample line providing flow to the Gross Failed Fuel Sample Cooler. The gauge is located on the 67'-6" elevation of the PAB adjacent to the gross failed fuel detector. This gauge can be manually isolated via valves SP-650A and/or SP-650B.

2.4.6 Loop Flow Instruments

Three flow transmitters are installed in the intermediate leg of each RCS loop. Their primary function is to indicate whether a reduction in loop flow rate has occurred rather than to provide an accurate measure of loop flow rate.

The flow transmitters shown in Figure 1.1-12 are differential pressure cells that develop a signal proportional to the flow rate. Unlike typical flow detectors that use a flow-restricting device to produce a pressure drop, the RCS loop flow transmitters are installed on the elbow in RCS piping at the RCP suction. As the coolant flows through the elbow, it exerts a higher pressure on the outer curvature of the elbow than it does on the inner curvature. This pressure difference is a result of the difference in centrifugal force on the fluid between the inner and outer curvature. The expected accuracy of this method of flow detection is ±10%. Density compensation is not required because of the minimal variations in the intermediate leg coolant density as a function of the unit power.

Three low pressure root lines are used to ensure that a single failure cannot prevent the loop flow instrumentation from generating a loop low flow signal. If a low pressure root line ruptures, the pressure in the line decreases. This would cause the differential pressure to increase, indicating a high loop flow rate. This is a non-conservative failure. With three independent low pressure root lines, failure of one line leaves two other transmitters available to detect an actual low flow condition.

Only one high pressure root line is required because its failure (rupture) would cause all three differential pressure detectors to sense a low flow. Thus, the loop flow instruments fail safe (produce a loop low flow signal) if the single high pressure root line fails. Using only one high pressure line reduces the number of penetrations in the RCS.

Each flow tap is provided with a manual root isolation valve (HP and all 3 LP taps). For Loop 31, these are RC-513 (HP tap) and RC-514A, RC-514B, and RC514C (LP taps).

Figure 1.1-26 presents the three flow instruments for RCS loop 31 and is typical for all loops. Each flow transmitter supplies

- A flow indicator (FI-414, FI-415, or FI-416) on panel SAF in the control room (range 0-120%),
- A computer input (CI), and
- A low flow comparator bistable (FC-414, FC-415, or FC-416).
    - ♦ At 93% of normal flow on one of the three sensors, the comparator generates a low flow trip signal and an alarm on panel SAF.
    - ♦ If two of the three sensors detect 93% of normal flow, the coincidence logic gate transmits a loop low flow signal. This

2.2.2 P-7 Permissive (Figure 28-15)

The P-7 permissive is used to block the high pressurizer level, low pressurizer pressure reactor trips, reactor coolant low flow and undervoltage reactor trip signals to the reactor protection system. The P-7 permissive is activated by a bistable circuit indicating less than 10% power as measured by both turbine first stage pressure detectors and 3/4 power range channels. The power range input is supplied by the P-10 permissive. A white "POWER BELOW P-7" lamp illuminates on the control room FBF panel while the P-7 permissive is active and extinguishes when reactor power and/or turbine power are >10%.

2.2.3 P-8 Permissive (Figure 28-14)

The P-8 permissive blocks the automatic reactor trip on low flow in one loop if power is below 35% at the time one RCP is lost. The permissive also blocks a reactor trip due to a turbine trip when power is below 35%. A white "POWER BELOW P-8" lamp illuminates on the control room FBF panel when the P-8 permissive is active and extinguishes when reactor power is above 35%.

2.2.4 P-10 Permissive (Figure 28-14)

The P-10 permissive blocks the intermediate range channel and low power range channel trips during an approach to power. It is also used to backup the P-6 permissive to block the Source Range instrumentation and is one of the inputs to the P-7 permissive.

When 2 of 4 power range channels indicate greater than 8.5% power the P-10 permissive is activated and a white "POWER ABOVE P-10" lamp illuminates. Once the P-10 lamp is lit, the low power and intermediate range hi flux trips may be manually blocked as described in the sections for those trips.

The P-10 permissive and associated manual blocks are automatically reinstated if power falls below 8.5% on 3/4 Power Range channels.

2.2.5 Low Power Auto Rod Withdrawal Block

Automatic control rod withdrawal is blocked until turbine power, as sensed by PT-412A (turbine first stage pressure), is greater than 15%. Automatic control rod insertion is not blocked. A white "LOW PWR AUTO ROD WITHDRAWAL BLOCK" light, on the FBF panel, is illuminated when the permissive is active and extinguishes when turbine power is >15%.

- Power > P-8 - 4 Reactor Coolant Pumps
- Power > P-7 - 3 Reactor Coolant Pumps
- Power < P-7 - no pumps are required

All reactor coolant flow trips are blocked below 10% power by permissive P-7 because natural circulation is capable of cooling the core below 10% power. Administratively, all four reactor coolant pumps are required to be in operation any time the reactor trip breakers are closed (T.S. is less restrictive).

Three flow measuring circuits monitor each reactor coolant loop and are arranged in a two out of three logic. A low flow trip will occur when any two of three channels indicate that flow has decreased to 93% of normal full flow. This trip signal is used in two logic circuits. The first circuit is used to trip the reactor if power is > P-8 and flow is lost in any one loop. P-8 is activated when reactor power increases to 35%. The second logic circuit is used to trip the reactor if power is greater than or equal to 10% and flow is low in 2 or more loops. P-7 is activated when power is less than 10% blocking the low flow trips.

A single channel tripping would annunciate the "REACTOR COOLANT LOOP 31 (32,33,34) LOW FLOW CHANNEL TRIP" Alarm on the SAF panel in the Control Room.

The RCP breaker open signal is derived from a breaker auxiliary "a" contact to actuate a low flow trip signal. This trip signal is provided to anticipate probable plant transients and to avoid the resultant thermal transient. The trip signal is sent to a one out of two logic device that senses breaker position and loop flow. A breaker open signal will initiate a signal that is sent to two logic circuits to compare the flow combination with the reactor power level. These circuits are the ones described above. With power above 35% and P-8 inactive one RCP breaker open signal causes a trip; with power below P-8 and >10% AND P-7 inactive it takes two RCP trip signals to cause the trip.

The RCP undervoltage trip signal provides protection following a complete loss of power. The RCP's are powered from 6.9KV buses 1 through 4. A decrease in voltage to 75% of nominal voltage, as sensed by the undervoltage relays, initiates the signal. The four bus undervoltage relay signals are sent to a two out of four logic circuit. It requires two bus undervoltage signals to generate the undervoltage reactor trip. This signal indicates a probable loss of power and protects the core from DNB. Permissive P-7 blocks this trip below 10% power.

A sustained undervoltage condition for .5 seconds, overcurrent, underfrequency on two out of four 6.9 kV busses (1-4) or a ground condition trips the RCP breaker using a one out of four logic. These

QuestionId	19403	ExamType	ILO	ExamDate	12/11/2000
AbbrevLocName	Kewaunee, Unit 1	NSSSVendor	#Name?	NSSSType	PWR

QuestionStem

Given the following conditions:

- The plant is operating at 18% power
- The high pressure tap to RCS flow instrument FT-411 on loop A fails

What is the resulting plant condition, if NO operator action is taken?

QuestionCommen

CognitiveLevel		ExamLevel	R	RefMaterial	N	ParentQuestionId	

Answer

All loop A flow indicators will read low, and the reactor trip is generated on RCS loop low flow.

Distract1

All loop A flow indicators will read low, but the reactor trip is generated on low PRZR pressure.

Distract1Co

Distract2

Only FI-411 RCS flow indication will read low, and the reactor trip is generated on RCS loop low flow.

Distract2Co

Distract3

Only FI-411 RCS flow indication will read low, but the reactor trip is generated on low PRZR pressure.

Distract3Co

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 2	Group #	1	
	K/A #	000008AA2.20	
OK	Importance Rating	3.4	

Ability to determine and interpret the effect of an open PORV on code safety, based on observation of plant parameters as they apply to the Pressurizer Vapor Space Accident

Proposed Question: Common 2

The following conditions are observed about 3 minutes after an automatic safety injection:

- Core exit T/Cs 540°F
- Pressurizer level 58% and increasing
- RCS pressure 1100 psig and decreasing
- Containment pressure 0 psig

Based on these indications, it is likely that a;

A. small break LOCA has occurred outside the containment.

B. steam line break has occurred outside containment.

C. steam generator tube rupture has occurred.

D. a pressurizer PORV is stuck open.

Proposed Answer:
D. pressurizer PORV is stuck open.

Explanation (Optional):

Technical Reference(s): SD-1.4 (Attach if not previously

provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSPZR E6a (As available)

Question Source: Bank # INPO 26069
Modified Bank # (Note changes or attach parent)
New

Question History: 9/01/2003 Prairie Island 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3,7,14
55.43 5

Comments:

WS #2

- if the RCS pressure increases to greater than 2335, then the operator must ensure the PORVs are unisolated (block valves open) and have opened.

Pressure decreases below the normal control band if

- a PRZR spray valve fails open (or fails to close),
- a PRZR PORV fails open (or fails to close)
- all PRZR heaters fail off
- the auxiliary spray valve fails open.

The first two possible causes (spray valve or PORV failure) can cause a rapid reduction in pressure. Without rapid operator response, a reactor trip and a safety injection on low pressurizer pressure occur.

If the operator has time to respond, ONOP-RCS-2 directs the operator to check for failed components based on their relative effect on PRZR pressure:

- ensure the PORVs are closed. If a PORV does not close, then its block valve is closed and the fuses to the PORV's solenoid circuit are removed. Removing the PORV solenoid circuit fuses causes the PORV to fail closed;
- ensure both spray valves are closed. If a spray valve is open in AUTO and it should be closed, it is closed in MANUAL. If it cannot be closed in MANUAL, then the valve controller's power fuses are removed. Removing the power fuses should cause the valve to fail closed. If a spray valve is still open, the RCPs (which provide the driving force for spray) must be tripped. Tripping RCPs requires that the Reactor be tripped first;
- If pressure is less than 2185 psig (and the PORVs and spray valves are not the cause of the event, or have been closed), ensure all the backup heaters have energized to restore RCS pressure. If the backup heaters can not be energized from the control room, then the 31 backup heaters can be operated from the Local Pressurizer Pressure and Level Control Panel in the PAB.
- Ensure that auxiliary spray valve AOV-212 is closed and is not leaking.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 3	Group #	1	
	K/A #	000009EK2.03	
OK	Importance Rating	3.0	

Knowledge of the interrelations between the small break LOCA and the SGs

Proposed Question: Common 3

The following plant conditions exist:

- A reactor trip with SI has occurred.
- The crew transitioned from E-0, Reactor Trip or Safety Injection, to FR-H.1, Loss of Secondary Heat Sink, based on valid red path condition on the heat sink CSF.
- RCS pressure is 700 psig and slowly decreasing.
- All S/G pressures are approximately 950 psig and stable.

Which of the following summarizes plant conditions and what procedure should be implemented?

A. Because S/Gs are the sole heat sink, a transition to E-1, Loss of Reactor or Secondary Coolant, is made to minimize coolant loss and restore S/G levels to normal band.

B. Heat transfer in the RCS during this event is such that the S/Gs are currently NOT functioning as a heat sink and therefore NOT required. Return to E-0 then transition to E-1, Loss of Reactor or Secondary Coolant.

C. Heat transfer in the RCS during this event is such that the S/Gs are currently NOT functioning as a heat sink. Remain in FR-H.1 to restore S/G levels to normal band.

D. Remain in FR-H.1 until feed is restored then transition to E-1 where a depressurization of the secondary is prescribed to increase the heat transfer between the RCS and S/Gs.

Proposed Answer:

B. Heat transfer in the RCS during this casualty is such that the S/Gs are currently NOT functioning as a heat sink and therefore NOT required. Return to E-0 then transition to E-1, Loss of Reactor or Secondary Coolant.

Explanation (Optional):

Technical Reference(s): FR-H.1 bases, step 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRH 7 (As available)

Question Source: Bank # INPO 24717
Modified Bank # (Note changes or attach parent)
New

Question History: 5/30/2003 Seabrook 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5,7
55.43

Comments:

WS #3

STEP DESCRIPTION TABLE FOR FR-H.1 Step 1

STEP: Check If Secondary Heat Sink Is Required

PURPOSE: To check if a secondary (SG) heat sink is required for heat removal

BASIS:

Before implementing actions to restore flow to the steam generators, the operator should check if secondary heat sink is required. For larger LOCA break sizes, the RCS will depressurize below the intact steam generator pressures. The steam generators no longer function as a heat sink and the core decay heat is removed by the RCS break flow. For this range of LOCA break sizes, the secondary heat sink is not required and actions to restore secondary heat sink are not necessary. For these cases, the operator returns to the guideline and step in effect.

Since Step 8 directs the operator to return to Step 1 if the loss of secondary heat sink parameters are not exceeded, break sizes that take longer to depressurize the RCS will be detected on subsequent passes through Step 1.

If RCS temperature is low enough to place the RHR System in service, then the RHR System is an alternate heat sink to the secondary system. Therefore, an attempt is made to place the RHR System in service in parallel to the attempts to reestablish feedwater flow. RCS pressure must be below normal RHR System pressure limits.

ACTIONS:

- Determine if RCS pressure is greater than any nonfaulted SG pressure
- Determine if RCS temperature is greater than (F.06)°F [(F.07)°F for adverse containment]
- Determine if adequate cooling is established with RHR System
- Try to place RHR System in service while continuing with guideline
- Return to guideline and step in effect

INSTRUMENTATION:

- RCS pressure indication
- SG pressure indication
- RCS temperature indication
- Plant specific RHR System instrumentation including valve position and pump status indications

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	<u>CHECK AFW Status:</u>	
	a. VERIFY total AFW flow - GREATER THAN 365 GPM	a. PERFORM the following: 1) Manually START available pump(s). 2) ALIGN valves as required. 3) <u>IF</u> cutback controller is malfunctioning, <u>THEN</u> ATTEMPT manual control. 4) <u>IF</u> all SG NR levels are less than 9% [14%] <u>AND</u> total AFW flow can <u>NOT</u> be maintained greater than 365 gpm, <u>THEN</u> GO To FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
	b. CONTROL feed flow to maintain SG NR levels between 9% [14%] and 50%	

<u>CAUTION</u>

STARTING OF EQUIPMENT MUST BE COORDINATED WITH ALL CONTROL ROOM OPERATORS TO ENSURE THAT TWO COMPONENTS ARE <u>NOT</u> STARTED AT THE SAME TIME ON THE SAME POWER SUPPLY.

* *

6. * <u>DIRECT BOP Operator to PERFORM RO-1, BOP OPERATOR ACTIONS DURING USE OF EOPS</u> *

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- IF TOTAL FEEDFLOW IS LESS THAN 365 GPM DUE TO OPERATOR ACTION, THIS PROCEDURE SHOULD NOT BE PERFORMED.
- FEEDFLOW SHOULD NOT BE REESTABLISHED TO ANY FAULTED SG IF A NON-FAULTED SG IS AVAILABLE.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	DETERMINE If Secondary Heat Sink Is Required:	
	a. CHECK RCS pressure - GREATER THAN ANY NON-FAULTED SG PRESSURE	a. RETURN To Procedure and Step in effect.
	b. CHECK RCS hot leg temperatures - ANY GREATER THAN 350° F	b. PERFORM the following: 1) TRY to place RHR cooling in service while continuing with this procedure: • REFER TO SOP-RHR-1, RESIDUAL HEAT REMOVAL SYSTEM. 2) WHEN RCS temperature is stable or decreasing on RHR cooling, THEN RETURN To Procedure and Step in effect.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 5	Group #	1	
	K/A #	000015/17G2.1.28	
OK	Importance Rating	3.2	

Knowledge of the purpose and function of major system components and controls

Proposed Question: Common 4

When a RCP is stopped IAW 3-AOP-RCP-1, RCP MALFUNCTION, due to high #1 Seal leakoff flow, the respective Seal Leakoff Isolation Valve, is closed after stopping the pump.

Which one of the following correctly describes the reason for closing the respective Seal Leakoff Isolation Valve?

A. Prevent excessive back pressure from interfering with leakoff from the operating RCP's.

B. Prevent over-pressurization of the return line and a possible LOCA Outside Containment.

C. Prevent thermal shock to the in-service CCW Heat Exchanger.

D. Reduce RCS inventory loss by directing all #1 seal leakoff to #2 Seal.

Proposed Answer:

D. Reduce RCS inventory loss by directing all #1 seal leakoff to #2 Seal.

Explanation (Optional):

Technical Reference(s): 3-AOP-RCP-1 step 13 bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSRCP H
I3LP-ILO-AOPRCP K (As available)

Question Source: Bank # INPO 23133
Modified Bank # (Note changes or attach parent)
New

Question History: Salem Unit 1 11/4/2002

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS #5

INSTRUCTOR LESSON PLAN

PRESENTATION DATA	Activity/Notes
2) Before performance of E-0 IOAs	
c. INITIATE E-0	
1) Procedure remains open and in effect	
2) Additional actions may be required to mitigate the event	
a) Close seal return valve	
(1) Backpressure on #2 seal	
(2) Stop flow of hot water to VCT	
(3) Stop flow of hot water over thermal barrier heat exchanger	Boiling in CCW system
(4) Stop Flow of hot water over #1 & #2 seals	Prevent further degradation
3. IAAT affected RCP has stopped rotating	Step 4.13 Step 4.23
▪ RCPs operating and flow 20 – 30 % affected loop	
▪ All RCPs secured and flow 0%	
a. Close Seal return isolation valve (261A-D)	
1) Backpressure on #2 seal	
2) Stop flow of hot water to VCT	
3) Stop flow of hot water over thermal barrier heat exchanger	Boiling in CCW system
4) Stop Flow of hot water over #1 & #2 seals	Prevent further degradation
b. Close affected spray valve if 33 or 34 RCP	
1) Prevent short cycling spray flow from unaffected loop	
4. IATT #1 seal return flow is < 0.84 gpm check conditions to trip reactor and stop affected RCP	Step 4.17

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 8	Group #	1	
	K/A #	000027AK2.03	
OK	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Pressure Control Malfunction and controllers and positioners

Proposed Question: Common 5

The plant is operating at 100% power with all systems and controls in a normal lineup.

Which ONE of the following could cause a Reactor Trip AND Safety Injection actuation? (Assume NO operator action.)

A. TE-433B, Loop 34 Cold Leg (Channel 4), fails LOW.

B. PT-455, Pressurizer Pressure (Channel 1), fails HIGH

C. A trip of both Main Feed Pumps.

D. Direct Trip from Buchanan.

Proposed Answer:

B. PT-455, Pressurizer Pressure (Channel 1), fails HIGH

Explanation (Optional):

Technical Reference(s): SD 1.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICPZPC E-5.a (As available)
I3LP-ILO-AOPINT I.a.

Question Source: Bank # INPO 21540
Modified Bank # (Note changes or attach parent)
New

Question History: 9/06/2002 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS #8

high level reactor trip bistable. Once these actions are completed, maintenance repairs the instrument.

3.2.2 PRZR Pressure Channel Failure

PRZR pressure instrument failures are addressed in ONOP-RPC-1, Instrument Failures. The following discussion assumes all systems are aligned for normal operations.

If any pressurizer pressure channel, PT-455, PT-456, PT-457, or PT-474, fails high, the affected channel indicates full scale and the Pressurizer High Pressure Channel Trip alarm is triggered on panel SAF (See Figure 1.4-22). If the failed channel is being used for alarm train (PT-456 or PT-474):

- PCV-456 (PORV) receives an open signal, however it does not open because the open permissive interlock is not satisfied (PT-457 is not greater than 2335 psig), and
- Pressurizer High Pressure alarm is triggered on panel SAF.

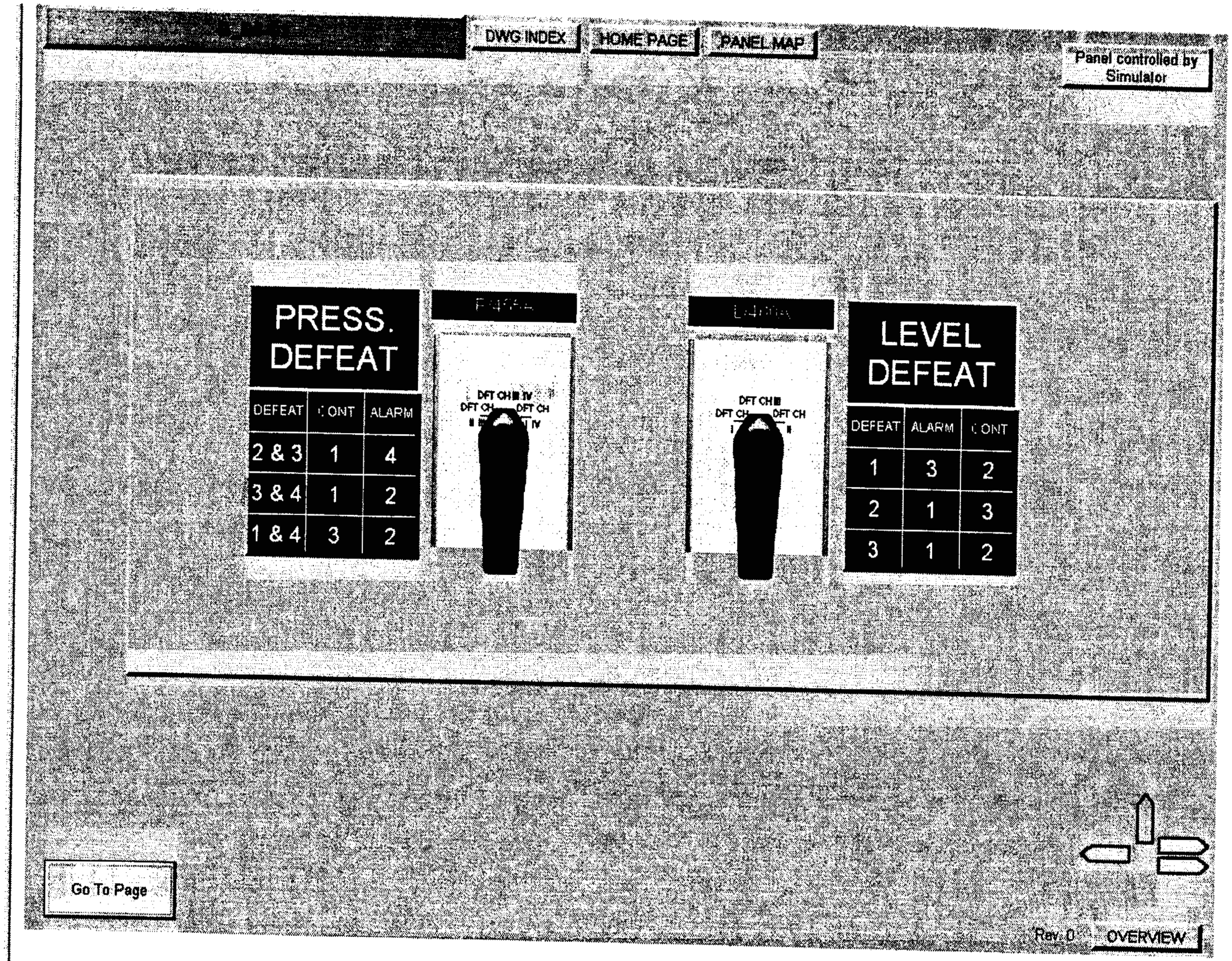
If the failed channel is being used as the input to the control train (PT-455 or PT-457), pressure controller PC-455K senses a large deviation above the setpoint pressure and outputs a large P_{ERROR} signal:

- Pressurizer High Pressure alarm panel SAF,
- Control Heater Group are fully de-energized,
- both spray valves go full open causing a rapid lowering of actual PRZR pressure,
- RC-PCV-455C (PORV) receives an open signal, however it does not open because permissive interlock is not satisfied (PT-474 pressure is not above 2335 psig),

If operator action is not taken, the decrease in pressure causes protective actions:

- a low pressure reactor trip when two of the remaining three pressure channels decrease to the trip setpoint;
- a safety injection actuation when two the operable pressure SI input channels reach the 1720 psig setpoint.

If a PRZR pressure channel fails low, the affected channel indicates down scale. The Pressurizer Low Pressure Channel Trip and Pressurizer Low Pressure (SI) Channel Trip alarms are triggered on panel SAF (Part of Reactor Protection and Safeguards not shown on


DWG INDEX
HOME PAGE
PANEL MAP
Panel controlled by Simulator
PRESS. DEFEAT
DEFEAT CONT ALARM
2 & 3 1 4
3 & 4 1 2
1 & 4 3 2
DFT CH III IV
DFT CH
DFT CH
LEVEL DEFEAT
DEFEAT ALARM CONT
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DFT CH III
DFT CH
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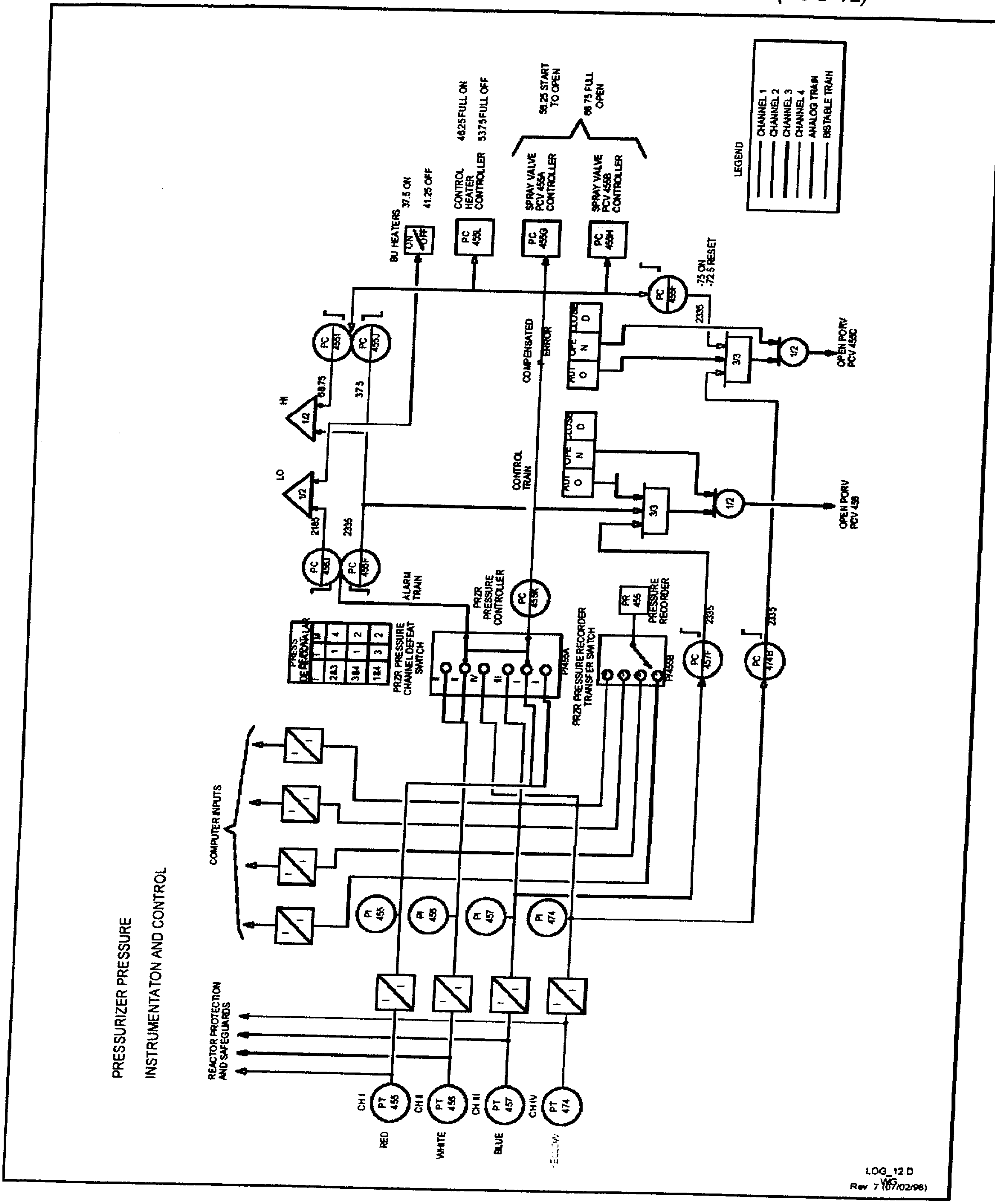
OVERVIEW

Figure 1.4-22: Pressurizer Pressure Instrumentation and Control (LOG-12)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 9	Group #	1	
	K/A #	000029EK1.01	
OK	Importance Rating	2.8	

Knowledge of the operational implications of reactor nucleonics and thermo-hydraulics behavior as they apply to the ATWS

Proposed Question: Common 6

During an ATWS event, the fuel cladding fission product barrier is severely challenged.

Which ONE of the following conditions is the mechanism which causes the fuel/cladding challenge?

A. Heat generated from the Zr-H_2O reaction.

B. Excessive radial flux distribution.

C. High RCS pressure caused by high temperature.

D. Fuel overheating from DNBR limits being exceeded.

Proposed Answer:
D. Fuel overheating from DNBR limits being exceeded.

Explanation (Optional):

Technical Reference(s): EOP FR-S.1 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRS 3 (As available)

Question Source: Bank # INPO 20212
Modified Bank # (Note changes or attach parent)
New

Question History: 9/10/2001 Cook 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 1,7,14
55.43 6

Comments:

ws #9

2.3 ATWS Events

ATWS events are postulated to be initiated from Condition II transients. Commonly termed "Anticipated Transients" to distinguish them from the more severe, lower probability Condition III and IV transients, they include:

1) Uncontrolled RCCA Bank Withdrawal
2) Uncontrolled RCCA Bank Misalignment
3) Partial Loss of Forced Reactor Coolant Flow
4) Loss of Load and/or Turbine Trip
5) Loss of Normal Feedwater
6) Station Blackout (Loss of Offsite Power)
7) Accidental RCS Depressurization

The basic design criteria for Condition II events require that they be tolerated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage is not expected for Conditon II events, although a small number of fuel rods may experience limited damage. These are within the capability of the plant clean-up systems.

A common characteristic of these events is a power generation-power removal mismatch leading to temperature excursions of the RCS. Some are characterized by increasing RCS pressures and others by RCS depressurization. It is usual to evaluate the core performance in terms of changes in the DNB ratio (or DNBR). The design duty cycle and SAR analyses report the results of these events in terms of DNBR changes. These results are used, in part, to establish the set points for the reactor protection system.

ATWS events are postulated to initiate from the Condition II transients, except that the reactor protection system is assumed to malfunction in a manner to preclude rod drop into the core. Several sets of analyses have been

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 11	Group #	1	
	K/A #	W/E12G2.2.34	
OK	Importance Rating	2.8	

Knowledge of the process for determining the internal and external effects on core reactivity

Proposed Question: Common 7

Given the following:

- Operation at 100% power at EOL
- Unit 3 experienced a large steam line break on the common steam header.
- A Reactor Trip and Safety Injection occurred.
- NONE of the MSIV's closed automatically or manually.

Which one of the following describes the response of core reactivity following the Reactor Trip and Safety Injection?

A. Reactivity DECREASED due to the RCS cooldown then INCREASED due to the Safety Injection.

B. Reactivity INCREASED due to the RCS cooldown then DECREASED due to the Safety Injection.

C. Reactivity DECREASED throughout the whole transient.

D. Reactivity INCREASED throughout the whole transient.

Proposed Answer:

B. Reactivity INCREASED due to the RCS cooldown then DECREASED due to the Safety Injection.

Explanation (Optional):

Technical Reference(s): EOP ECA-2.1 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPE20 2 (As available)

Question Source: Bank # INPO 22527
Modified Bank # (Note changes or attach parent)
New

Question History: 10/1/2002 Diablo Canyon 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 1
55.43 6

Comments:

Intermediate Size Break

A break is assumed that will result in all loops experiencing decreasing steam pressure and increasing steam load for which control systems are unable to compensate. Steam generator water level and primary average temperature will slowly decrease and the control rods will commence stepping out of the core in an attempt to maintain nominal primary system average temperature. Also, due to the decreasing temperature, a primary pressure decrease occurs. This trend will continue until either the operator manually trips the reactor or a trip setpoint is reached on overpower ΔT, low steamline pressure, or low pressurizer pressure. In any case, signals would eventually be generated for reactor trip, turbine trip, safety injection, feedwater isolation, steamline isolation, and auxiliary feedwater initiation. With a failure to restore integrity to any steam generator (e.g., all MSIVs fail to close), all steam generators would continue to blowdown to atmospheric pressure resulting in a continued decrease in primary system temperature and pressure. As the primary system temperature drops, the heat transfer to the steam generators and the primary system cooldown rate will be reduced. This trend will continue to the point where the primary system water volume shrinkage (caused by the cooldown) is overcome by the safety injection system flowrate. This results in the primary system pressure and pressurizer level restoration. Feed flow is then adjusted to control RCS temperatures.

Large Size Break

For the double-ended break, an immediate decrease in steamline pressure to the low steamline pressure setpoint (0.5-10 seconds) results in safety injection, feedline isolation, reactor trip, turbine trip and auxiliary feedwater initiation signals. With the failure to restore secondary pressure boundary integrity to any steam generator (i.e., common failure of the main steam isolation valves), a rapid, extensive primary system cooldown and depressurizaton occurs. As the primary system temperature drops, the heat transfer to the steam generators and the primary system cooldown rate will be reduced. This trend will continue to the point where the primary system water volume shrinkage (caused by the cooldown) is overcome by the safety injection system flowrate. This results in the primary system pressure and pressurizer level restoration. Depending on the initial conditions of the systems and the size and location of the break, one of two conditions will be reached on the blowdown. The first is when the steam generator blowdowns are essentially completed and further cooldown of the primary system is controlled by the auxiliary feedwater flow. The second is when the primary temperature is reduced so far that the heat transfer to the steam generators matches the heat generation in the primary system which results in a stabilized primary temperature.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 4	Group #	1	
	K/A #	000011EA2.14	
OK	Importance Rating	3.6	

Ability to determine or interpret the actions to be taken if limits for PTS are violated as they apply to the Large Break LOCA

Proposed Question: Common 8

Given the following conditions:

- A LOCA had occurred from HOT SHUTDOWN conditions 30 minutes ago
- RCS pressure is 125 psig
- RCS Core Exit TCs read 380°F
- RCS Cold Leg temperatures are all 220°F
- 31 SI Pump is running providing 325 gpm flow
- 31 RHR Pump is running providing 1150 gpm flow

What is the appropriate action taken in response to the above conditions?

Entry into FR-P.1, Response to Pressurized Thermal Shock Condition, is ...

A. NOT required since RCS pressure is below 350 psig.

B. made but NO actions are implemented before returning to procedure in effect.

C. made and a RCS temperature soak for a ONE hour period will be completed.

D. made and cooldown will continue within a limit of 50°F in any 60 minute period.

Proposed Answer:

B. made but NO actions are implemented before returning to procedure in effect.

Explanation (Optional):

Technical Reference(s): FR-P.1, step 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRP 12 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3,5,7
55.43 5

Comments:

WS#4

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	CHECK RCS Pressure - GREATER THAN 325 PSIG [650 PSIG]	IF RHR flow is greater than 300 gpm on each of two indicators, THEN RETURN To Procedure and Step in effect.

CAUTION

IF THE TURBINE-DRIVEN AFW PUMP IS THE ONLY AVAILABLE SOURCE OF FEED FLOW, THEN THE STEAM SUPPLY TO THE TURBINE-DRIVEN AFW PUMP MUST BE MAINTAINED FROM ONE SG.

NOTE

A faulted SG is any SG that is depressurizing in an uncontrolled manner or completely depressurized.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	CHECK RCS Cold Leg Temperatures - STABLE OR INCREASING	STOP RCS cooldown as follows: a. VERIFY SG atmospherics are closed. b. VERIFY condenser steam dump valves are closed. c. IF RHR System is in service, THEN STOP any cooldown from RHR System.

(STEP 2 CONTINUED ON NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 12	Group #	1	
	K/A #	000054AK1.01	
OK	Importance Rating	4.1	

Knowledge of the operational implications of the MFW line break depressurizes the SG as they apply to the Loss of Main Feedwater

Proposed Question: Common 9

Given the following plant conditions on Unit 3:

- Reactor power is 90%.
- RCS Tave is stable at 565°F on all 4 loops
- RCS pressure is stable at 2235 psig
- Containment Pressure is INCREASING
- 33 SG Feed Flow is pegged HIGH
- 33 SG Main FW Reg Valve is full OPEN.
- 33 SG pressure is STABLE
- 33 SG level is DECREASING

Which of the following events is in progress?

A. Main Feed Pump trip

B. Feed Flow Indicator failed HIGH.

C. Feed Line Break INSIDE Containment.

D. Main FW Reg Valve failed OPEN.

Proposed Answer:

C. Feed Line Break INSIDE Containment.

Explanation (Optional):

Technical Reference(s): ______________________ (Attach if not previously provided)

____

____

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPE20 2 (As available)

Question Source: Bank # INPO 19254
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: 10/20/2000 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8,10
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 13	Group #	1	
	K/A #	000055EK3.02	
OK	Importance Rating	4.3	

Knowledge of the reasons for actions contained in EOP for loss of offsite and onsite power as they apply to the Station Blackout

Proposed Question: Common 10

The following plant conditions exist:

- Unit 3 has experienced a loss of all AC Power due to severe weather conditions and failure of emergency diesel generators to start and supply safeguard buses.
- The operating crew is carrying out actions of ECA-0.0, Loss of All AC Power.
- Immediate actions have been completed and steps to restore power are in progress.
- The operators are at a point where they are to commence cooldown and depressurization of the steam generators.

Based on these conditions, which ONE of the following statements describes the reason why a secondary depressurization is directed?

A. To prevent a challenge to the Integrity Critical Safety Function Status Tree which is being monitored for implementation.

B. To minimize RCS inventory loss through the RCP seals which maximizes time to core uncovery.

C. To ensure the reactor remains subcritical and does not result in a restart accident.

D. To remove all the stored energy in the steam generators to prevent a secondary safety valve from lifting.

Proposed Answer:

B. To minimize RCS inventory loss through the RCP seals which maximizes time to core uncovery.

Explanation (Optional):

Technical Reference(s): ECA-0.0 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPC00 8 5804 (As available)

Question Source: Bank # INPO 20572
Modified Bank # (Note changes or attach parent)
New

Question History: 2/2/2002 Point Beach 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5,10
55.43 5

Comments:

WS#13

STEP DESCRIPTION TABLE FOR ECA-0.0 Step 16

STEP: Depressurize Intact SGs To (0.08) PSIG

PURPOSE: To depressurize the intact steam generators

BASIS:

Step 16 depressurizes the intact SGs, thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss. The advantages to performing this action, as well as restrictions that apply during the action, are detailed in Subsection 2.3.

During SG depressurization, SG level must be maintained above the top of the SG U-tubes in at least one SG. Maintaining the U-tubes covered in at least one SG will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. Step 16a requires that SG level be in the narrow range in at least one SG before SG depressurization is initiated in Step 16b. If level is not in the narrow range in at least one SG, RNO 16a instructs the operator to maintain maximum AFW flow until narrow range level is established in one SG. When narrow range level is established, SG depressurization can be started or continued via Step 16b.

Step 16b instructs the operator to reduce SG pressures by depressurizing the intact SGs. Depressurization should be accomplished by opening the PORVs on the intact SGs to establish a maximum steam dump rate, consistent with plant specific constraints. The step is structured assuming that the operator can open and control SG PORVs from the control room. This structure assumes that the PORVs are air-operated and have dc control power and pneumatic power (i.e., either air reservoirs or nitrogen bottles) available. Some plants may not have the capability to open the SG PORVs from the control room. These plants should evaluate their capability to accomplish this step locally via PORV handwheels. Such an evaluation should consider accessibility and communications necessary to accomplish local PORV operation.

Once depressurization is initiated, maintenance of a specified rate is not critical. The depressurization rate should be sufficiently fast to expeditiously reduce SG pressures, but not so fast that SG pressures cannot be controlled. It is important that the depressurization not reduce SG pressures in an uncontrolled manner that undershoots the pressure limit, thus permitting potential introduction of nitrogen from the accumulators into the RCS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 101	Group #	1	
	K/A #	000026AA2.01	
	Importance Rating	2.9	

Ability to determine and interpret the location of a leak in the CCWS as they apply to the Loss of Component Cooling Water

Proposed Question: Common 11

The operators are responding to a CCW leak IAW 3-AOP-CCW-1, Loss of Component Cooling Water. Makeup to 31 and 32 CCW Surge Tanks has been established. Both surge tanks continue to decrease slowly. After splitting CCW Headers, 31CCW Surge Tank is stable and 32 CCW Surge Tank continues to decrease. Which of the following sets of components could be the source of the CCW leak?

A. RCP Motor cooling, 31 SI Pump or Spent Fuel Pit Heat Exchanger

B. 32 Waste Gas Compressor, Charging Pumps or 31 Aux Component Cooling Pump

C. CVCS Non-reg Heat Exchanger, RCP Motor cooling or Seal Water Return Heat Exchanger

D. Seal Water Return Heat Exchanger, 31 RHR Heat Exchanger or Reactor Vessel Support Pads

Proposed Answer:

C. CVCS Non-reg Heat Exchanger, RCP Motor cooling or Seal Water Return Heat Exchanger

Explanation (Optional):

A. 31 SI Pump and SFP HX on 31 Header
B. 32 WGC on 31 Header
C. Correct
D. 31 RHR HX on 31 Header

Technical Reference(s): SD-4.1 Figure 4.1.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CCW001 0001 (As available)

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content:
55.41
55.43 5

Comments:

WS 101

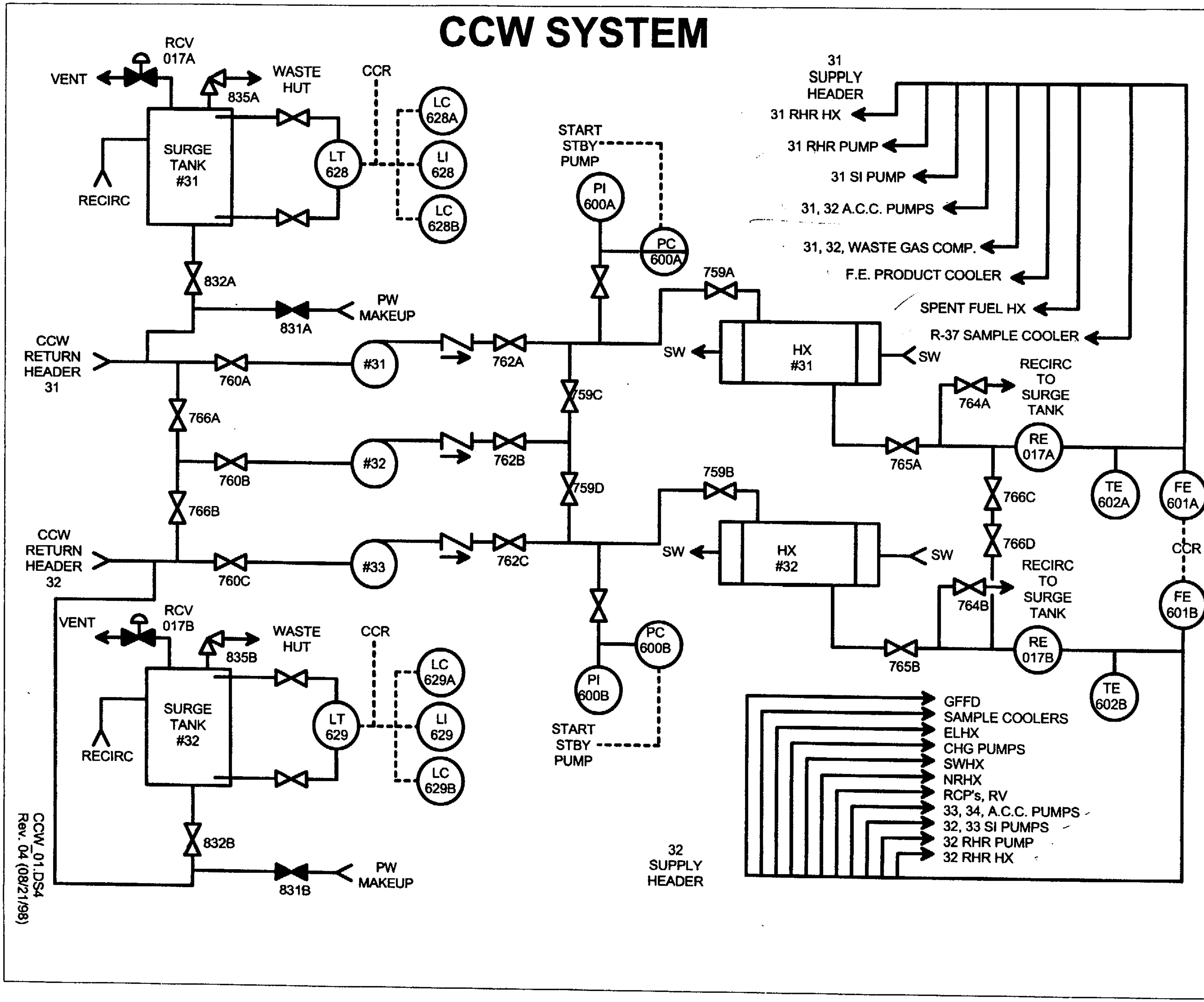


Figure 4.1-1: CCW SYSTEM (CCW_01.DS4)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 14	Group #	1	
	K/A #	000056AK1.01	
OK	Importance Rating	3.7	

Knowledge of the operational implications of the principles of cooling by natural convection as they apply to Loss of Offsite Power

Proposed Question: Common 12

Unit 3 was operating at 100% power when a reactor trip occurred due to a loss of offsite power. The operators completed the actions of ES-0.1, Reactor Trip Response and have transitioned to ES-0.2, Natural Circulation Cooldown, where they are initiating a natural circulation cooldown.

At the onset of the natural circulation cooldown, which ONE of the following processes will remove the MOST heat from the Reactor Vessel HEAD?

A. The 25°F/hr natural circulation cooldown of the RCS.

B. Heat losses to ambient.

C. All CRDM fans running.

D. Upper head bypass flow.

Proposed Answer:
C. All CRDM fans running.

Explanation (Optional):

Technical Reference(s): ES-0.2 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective:	I3LP-ILO-EOPE00 7		(As available)
Question Source:	Bank #	INPO	20211
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	9/10/2001	Cook 1	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	8,10,14	
	55.43		

Comments:

initial upper head temperature for these plants was conservatively chosen as T_{HOT}. Other Westinghouse plants operate with sufficient flow from the upper downcomer to the upper head region to make the upper head fluid temperature equal to the cold leg fluid temperature (T_{COLD}). Both types of plants were analyzed.

Another parameter which affects void formation in the upper head region is the cooldown rate of the primary system. Natural circulation cooldown rates of 25°F/hr and 50°F/hr were analyzed for T_{HOT} and T_{COLD} plants, respectively.

A final parameter important in the formation of voids in the upper head is the heat removal rate from the upper head. The two primary means of heat loss are ambient heat losses and heat removal by the control rod drive mechanism (CRDM) fans. The effect of ambient heat losses through the reactor vessel on upper head temperature is small compared to the effect of the CRDM fans. The cooloff rate of the upper head due to ambient heat losses is less than 1°F/hr and was neglected in the analysis. However, metal heat addition to the upper head area from the reactor vessel and upper internals was taken into account.

The CRDM cooling system consists of fans which maintain a suitable atmosphere within the CRDM shroud to protect and prolong the life of the CRDM motors. The system induces cooler containment air into the CRDM shroud and exhausts warmed air through the fans. The CRDM fans remove 8 kw/drive train at full power. For a 4-loop plant with 57 full-length and 8 part-length rods, the CRDM fans remove 8 kw x (57 + 8) = 520 kw. For a 3 or 2-loop plant, multiply the total number of full-length plus part-length rods by 8 kw/rod to obtain the heat removal comparable to the 520 kw for 4-loop plants. The ratio of heat removal by the CRDM fans to the upper head total energy (or upper head volume) is essentially the same for 2 and 3-loop plants as for 4-loop plants. Thus, the head cooldown rates determined for 4-loop plants (21°F/hr at 600°F and 11°F/hr at 350°F) are applicable to 2 and 3-loop plants.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 15	Group #	1	
	K/A #	000065AA1.05	
OK	Importance Rating	3.3	

Ability to operate and / or monitor the RPS as it applies to the Loss of Instrument Air

Proposed Question: Common 13

Unit 3 is at 100% power. IF a rupture occurs in the Instrument Air system, you should monitor plant conditions and initiate a manual reactor trip if plant conditions approach any automatic reactor trip setpoint.

Which plant parameter is going to reach its automatic reactor trip setpoint FIRST for this event?

A. Steam generator level

B. Pressurizer level

C. Pressurizer pressure

D. RCS loop ΔT (OTΔT)

Proposed Answer:

A. Steam generator level

Explanation (Optional):

Technical Reference(s): 3-AOP-AIR-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-IA001 7 1799 (As available)

Question Source: Bank # INPO 26084
Modified Bank # (Note changes or attach parent)
New

Question History: 9/1/2003 Prairie Island 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

LOS #15

Attachment 1
Valves of Immediate Concern

CONDENSATE SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
CD-LCV-1128	HOTWELL MAKE-UP	Closed
CT-LCV-1158-1	CONDENSATE STORAGE TANK LO LEVEL ISOLATION VALVE	Closed
CT-LCV-1158-2	CONDENSATE STORAGE TANK LO LEVEL ISOLATION VALVE	Closed

MAIN FEEDWATER SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
BFD-FCV-417	31 S/G MAIN FW REG	Closed
BFD-FCV-427	32 S/G MAIN FW REG	
BFD-FCV-437	33 S/G MAIN FW REG	
BFD-FCV-447	34 S/G MAIN FW REG	
BFD-FCV-417L	31 SG BYPASS FW REG	Closed
BFD-FCV-427L	32 SG BYPASS FW REG	
BFD-FCV-437L	33 SG BYPASS FW REG	
BFD-FCV-447L	34 SG BYPASS FW REG	
BFR-FCV-1115	31 MBFP RECIRC VALVE	Open
BFR-FCV-1116	32 MBFP RECIRC VALVE	

SERVICE WATER SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
SWN-FCV-1176	DIESEL GENERATOR SERVICE WATER FLOW CONTROL VALVE	Open
SWN-FCV-1176A		
SWN-TCV-1103	CONTAINMENT TEMPERATURE CONTROL VALVE	Open
SWN-TCV-1104		
SWN-TCV-1105		
SWN-TCV-1113	31 & 32 IACC HEAT EXCHANGERS OUTLET TEMPERATURE CONTROL VALVE	Open

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 6	Group #	1	
	K/A #	000022AK1.01	
OK	Importance Rating	2.8	

Knowledge of the operational implications of consequences of thermal shock to RCP seals as it applies to Loss of Reactor Coolant Pump Makeup

Proposed Question: Common 14

Unit 3 has entered 3-AOP-CVCS-1, Chemical and Volume Control System Malfunction due to a loss of all charging. The procedure cautions the operator not to re-establish seal injection if seal injection temperature has reached 225°F until the plant has been placed in Mode 5.

Which ONE of the following is the reason for this action?

A. To minimize thermal shock of the RCP seals.

B. To prevent thermal barrier heat exchanger failure.

C. To minimize the pressurizer control transient.

D. To prevent cocking the RCP seal causing excessive leakage

Proposed Answer:

A. To minimize thermal shock of the RCP seals.

Explanation (Optional):

Technical Reference(s): 3AOP-CVCS-1, Note prior to step 4.37 (Attach if not previously provided)

I3LP-ILO-EOPC00 5976

SD 1.3

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CVC001 8.h. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8,10

55.43

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.36 Dispatch an operator to close the following: __CH-241A (RCP 31 Seal Injection Flow Control Valve) __CH-241B (RCP 32 Seal Injection Flow Control Valve) __CH-241C (RCP 33 Seal Injection Flow Control Valve) __CH-241D (RCP 34 Seal Injection Flow Control Valve)	

NOTE

If seal inlet temperature reached 225°F, seal injection will not be restored until plant is in MODE 5.

4.37 __Determine and correct cause for loss of charging and seal injection.	
4.38 __**PERFORM** Attachment 1 (Charging and Letdown Restoration) (Page 45) to restore charging and letdown.	
4.39 __**RETURN** to procedure and step in effect.	

• • • END • • •

affected pump. The result of a #3 Seal failure is a standpipe low level for the affected pump.

3.2.1 Emergency Shutdown

Immediate pump shutdown is required if any of the following conditions occur:

- A loss of BOTH seal injection AND CCW cooling to any RCP.
- The pump #1 Seal Delta P is less than 200 psig.
- RCP seal return flow is less than 0.2 gpm.
- RCP seal in temperature exceeds 225°F.
- RCP seal outlet temperature is greater than 235°F.
- RCP vibration is greater than 5 mils Frame or 20 mils shaft.
- RCP Motor Winding temperatures exceed 250°F.
- Either motor bearing temperature reaches 200°F. In the event CCW is lost, temperatures will exceed 200°F in one to two minutes.

The reactor is to be placed in hot shutdown via normal operating procedures and the affected RCP secured if any of the following conditions occur:

- Seal injection temperature reaches 150°F.
- Motor bearing oil reservoir level alarms accompanied by abnormal pump indications.

Restoration of seal injection after a loss of both CCW and seal injection must be done very slowly (1°F/min) to reduce thermal stresses on the RCP internals. A seal package inspection and evaluation should be completed prior to any restart of an affected RCP.

Lesson Plan I3LP-ILO-EOPC00

a. Hot RCS fluid flows up shaft and is not cooled by thermal barrier.

b. The seals are not designed for 500 deg F + water.

c. Rubber "O Ring" deforms, causing loss of sealing and increased leakage.

d. A volume of cool injection water exists in the RCP seal area which can sustain cooling for a few minutes.

e. Maximum leakage is postulated to be 300 gpm per RCP.

   (1) This is based on full delta P across the RCP labyrinth seal of 2200 psid.

f. Tests on seal packages indicate that expected leakage is much less than the maximum, and on the order of 16 gpm per RCP.

g. It is impossible to predict how long the seal will last on a loss of all seal cooling.

h. New seal materials increase time to seal failure.

   (1) Seal coating - silicon nitrate.

   (2) New High Temperature O-Rings

3. Ask the students what is the best indicator available of RCP seal response. (Answer: after action to isolate the RCS (ECA-0.0 step 3), PZR level is best indicator.)

4. Seal cooling restoration.

   a. Following restoration of AC power, it is desirable to keep the RCP seal cooling isolated to prevent thermal shock of the RCP seals. TP - 6

   b. The RCP should only be started if an extreme (red level) or severe (orange level) challenge to a Critical Safety Function is diagnosed via Status Tree monitoring and the operator is instructed to start an RCP in the associated Function Restoration Procedure.

Reducing RCS temperature reduces the thermal degradation of materials and thermal expansion effects that tend to degrade the seal system sealing capability and sealing life. Consequently, any actions to reduce RCS pressure and temperature during a loss of all ac power event are consistent with minimizing RCS inventory loss and maximizing time to core uncovery.

Benefits and Consequences of Restoring Seal Cooling

Following the restoration of ac power, the operator will have the capability to restore seal cooling by reestablishing seal injection flow or reestablishing thermal barrier cooling using the component cooling water system. Restoring seal cooling may have several benefits such as reducing seal leakage and preventing further damage to the seal components. However, Westinghouse has not performed an analysis of how the RCP seal package will react as the seals cool, fits contract, the shaft moves, etc., possibly with partially extruded O-rings. There may be a potential to make seal leakage worse by restoring seal cooling, depending on how it is done.

The RCP Vendor Manual identifies limits for reestablishing seal cooling to a hot seal package to prevent further damage due to thermal shock and to prevent warping of the RCP shaft due to uneven cooling. These limits are only intended for a loss of seal cooling of short enough duration that the seal package heatup is limited. Although the limits have been extrapolated for an extended loss of seal cooling event in the past, they have not been validated for such an event that is beyond the design basis of the RCP. Therefore, no specific conclusions may be taken from the RCP vendor manual guidance for reestablishing seal cooling following an extended loss of seal cooling event. The following provides a qualitative assessment that determines the most appropriate method of restoring seal cooling following an extended loss of all ac power event:

To minimize the potential for thermal shock of the seals and shaft warping, component cooling water can be established to the thermal barrier heat exchanger before seal injection is established. Note that since the loss of all ac power event is beyond the design basis of the plant, the performance of the CCW system has never been analyzed under these conditions. Establishing

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 16	Group #	1	
	K/A #	W/E04EA1.2	
OK	Importance Rating	3.6	

Ability to operate and / or monitor operating behavior characteristics of the facility as they apply to a LOCA Outside Containment

Proposed Question: Common 15

Unit 3 is in MODE 4 cooling down on RHR with the following plant conditions:

- RCS Temperature 340°F slowly lowering
- RCS pressure 300 psig lowering
- PZR level 42% lowering
- CNMT temperature 100°F
- R-27, Wide Range Plant Vent Gas Activity Monitor, went into ALARM
- SG levels 42% (31) 40% (32) 43% (33) 40% (34)
- SG pressures 115 psig (31) 115 psig (32) 115 psig (33) 115 psig (34)

What event is taking place?

A. A steam leak has occurred inside CNMT.

B. The Cold Overpressure system has actuated.

C. A LOCA has occurred on the suction of the RHR pump.

D. Letdown line pressure control valve, PCV-135, has failed open.

Proposed Answer:
C. A LOCA has occurred on the suction of the RHR pump.

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3SG-ILO-AOPRHR 1 (As available)

Question Source: Bank # INPO 19269
Modified Bank # (Note changes or attach parent)
New

Question History: 10/20/2000 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5,10
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 17	Group #	1	
	K/A #	W/E11EA2.1	
OK	Importance Rating	3.4	

Ability to determine and interpret the facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to Loss of Emergency Coolant Recirculation

Proposed Question: Common 16

During a LOCA, emergency coolant recirculation capability was lost, and ECA-1.1, Loss of Emergency Coolant Recirculation, is currently in progress. A RED path is identified on the CONTAINMENT status tree, and transition to FR-Z.1, Response to High Containment Pressure, is performed.

What procedure should be used to operate the containment spray pumps, and why?

A. ECA-1.1, because it provides for REDUCED containment spray.

B. ECA-1.1, because an ECA should be completed prior to transferring to a Function Restoration Procedure.

C. FR-Z.1 because it takes precedence over ECA-1.1.

D. FR-Z.1, because it provides for GREATER containment spray.

Proposed Answer:
A. ECA-1.1, because it provides for REDUCED containment spray.

Explanation (Optional):

Technical Reference(s): ECA-1.1, step 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRZ 7 (As available)

Question Source: Bank # INPO 22433
Modified Bank # (Note changes or attach parent)
New

Question History: 10/1/2002 Diablo Canyon 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5

Comments:

WS #17

Number:	Title:	Revision Number:
FR-Z.1	RESPONSE TO HIGH CONTAINMENT PRESSURE	10

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**DETERMINE Which Procedure Should Control Containment Spray:**	
	a. CHECK ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION - IN EFFECT	a. GO To Step 4.
	b. GO To Step 5	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**DETERMINE If Containment Spray Is Required:**	
	a. CHECK containment pressure - HAS INCREASED TO GREATER THAN 22 PSIG	a. RETURN To Procedure and Step in effect.
	b. CHECK spray system - ALIGNED FOR INJECTION	b. IF aligned for recirculation, THEN PERFORM the following: 1) ENSURE at least one spray recirc stop valve is open: • 889A • 889B 2) GO To Step 4.f.
	c. CHECK spray pumps discharge valves - OPEN • 866A • 866B	c. Manually OPEN valve(s).
	d. CHECK spray addition tank discharge valves - OPEN • 876A • 876B	d. IF NaOH addition is desired, THEN OPEN valve(s).
	e. CHECK spray pumps - RUNNING	e. START spray pump(s).
	f. CHECK containment isolation Phase B valves - CLOSED	f. Manually CLOSE valve(s).
	g. STOP all RCPs	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	<u>**VERIFY Containment FCU Status:**</u>	
	a. CHECK FCUs - ALL RUNNING	a. Manually START FCU(s).
	b. PLACE FCU damper control switches in - INCIDENT MODE position	
	c. CHECK FCU dampers for all FCUs - IN INCIDENT MODE POSITION • Dampers A/B - CLOSED (inlet dampers) • Damper C - CLOSED (bypass damper) • Damper D - OPEN (filter outlet damper)	
	d. PLACE control switches for 1104 <u>AND</u> 1105 to OPEN	
	e. CHECK Serv Wtr Cont Clg valves - OPEN • 1104 • 1105	

CAUTION: If ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray should be operated as directed in ECA-1.1 rather than step 3 below.

PURPOSE: To ensure containment spray pumps are operated as directed in ECA-1.1 instead of this guideline, if ECA-1.1 is in effect

BASIS:

This caution warns the operator that the operation of the containment spray pumps indicated in guideline ECA-1.1 takes precedence over that noted in Step 3 of this guideline. This guideline specifies maximum available heat removal system operability in order to reduce containment pressure. Guideline ECA-1.1 uses a less restrictive criteria, which permits reduced spray pump operation depending on RWST level, containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in guideline ECA-1.1 since recirculation flow to the RCS is not available and it is very important to conserve RWST water, if possible, by stopping containment spray pumps.

ACTIONS:

Determine if ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 18	Group #	1	
	K/A #	W/E05G2.4.20	
OK	Importance Rating	3.3	

Knowledge of operational implications of EOP warnings, cautions and notes

Proposed Question: Common 17

A Unit 3 Reactor Trip occurred after a 200 day continuous run at 100% power. Following the trip, all AFW flow was lost and the Team transitioned to FR-H.1, Loss of Secondary Heat Sink. Due to distractions caused by a pressure channel failure, bleed and feed steps were not initiated until WR S/G levels were all <10%.

Which one of the following correctly describes the general consequence of the delay?

A. Core uncovery will be more severe only if the PRT rupture disk fails, increasing the loss of mass, while ECCS flow is limited by RCS pressure.

B. Core uncovery will NOT occur as long as one PZR PORV is open and one charging pump is injecting prior to SG dryout.

C. Core uncovery will NOT occur as long as both PZR PORVs are open and two charging pumps are injecting prior to SG dryout.

D. Core uncovery will be more severe because RCS pressure will remain at a higher value for a longer time, limiting ECCS flow.

Proposed Answer:

D. Core uncovery will be more severe because RCS pressure will remain at a higher value for a longer time, limiting ECCS flow.

Explanation (Optional):

Technical Reference(s): FR-H.1 Bases (Attach if not previously

provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRH 2 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

2. DESCRIPTION

A loss of secondary heat sink can occur as a result of several different initiating events. Possibilities are a loss of main feedwater during power operation, a loss of offsite power, or any other scenario for which main feedwater is isolated or lost when the steam generators provide the main heat removal path. For these initiating transients a failure of the auxiliary feedwater (AFW) system to inject or a loss of AFW early in the cooldown, before RHR System operation can be established, could lead to a loss of secondary heat sink.

A loss of all feedwater transient is characterized by a depletion of secondary inventory and eventual degradation of secondary heat transfer capability. As secondary heat transfer capability degrades, a loss of secondary heat sink results and core decay heat generation will increase RCS temperature and pressure until the pressurizer power operated relief valves (PORV) or pressurizer safety valves open to relieve the increasing RCS pressure. At this point the opening and closing of the PORVs or safety valves will result in a loss of RCS inventory similar in nature to a small break loss of coolant accident. If operator action is not taken, the pressurizer PORVs or safety valves will continue to cycle open and closed at the valve setpoint pressure removing RCS inventory and a limited amount of core decay heat until eventually enough inventory will be lost to result in core uncovery.

The plant status upon entering this guideline will be a function of the initiating event. If the initiating event is a loss of main feedwater during power operation with AFW flow unavailable, or from any other anticipated transient resulting in reactor trip and main feedwater isolation or failure with AFW flow unavailable, the transient may not result in an automatic SI actuation. If the initiating event has resulted in a reactor trip due to primary depressurization (i.e., small LOCA, secondary break or steam generator tube rupture) with AFW flow unavailable, then SI should have been automatically initiated. However, the status of SI upon entering the guideline is not important to the actions that will be taken. Should it become necessary to establish a bleed and feed heat removal path (actuating SI and manually opening all pressurizer PORVs), then SI will be established.

KNOWLEDGE:

- The importance of establishing bleed and feed as an alternative heat sink to prevent core uncovery and inadequate core cooling.
- If PORV block valves are closed, they should be opened at this time unless they are closed to isolate a faulty PORV.
- When the RCPs are stopped due to loss of heat sink, RCS pressure and temperature are expected to increase slightly and stabilize below the PRZR PORV setpoint. RCS pressure and temperature will continue to be relatively constant until SG dryout occurs (approximately 20 - 30 minutes). At this point, the primary-to-secondary heat transfer rate degrades and the RCS begins to heat up and repressurize and will eventually result in the opening of the PRZR PORVs.

  This should not be confused with the onset of natural circulation in which the RCS pressure continues to increase after the RCPs are stopped and may reach the PRZR PORV setpoint. The key to determining if the RCS pressure rise is due to loss of heat sink or natural circulation is the loop delta-T. The loop delta-T is expected to be large for natural circulation and small for a loss of heat sink since there is no heat transfer to the secondary.

  Therefore, verifying a slowly increasing RCS pressure and temperature trend plus a large loop delta-T prior to the PORV opening confirms natural circulation whereas a relatively stable temperature and pressure and a small loop delta-T combined with SG wide range low level prior to the PORV opening confirms a loss of heat sink.

PLANT-SPECIFIC INFORMATION:

- (X.01) Parameter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy. Refer to Background Document for guideline FR-H.1.
- (X.02) Parameter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy and post accident transmitter errors. Refer to Background Document for guideline FR-H.1.
- Parameters (X.01) and (X.02) are described in subsection 2.2.4, Plant-Specific Symptoms for Loss of Heat Sink, of this background document. Parameters and setpoints in this CAUTION should be consistent with Step 8.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 19	Group #	2	
	K/A #	000001AK1.06	
OK	Importance Rating	4.0	

Knowledge of the operational implications of the relationship of reactivity and reactor power to rod movement as they apply to the continuous rod withdrawal

Proposed Question: Common 18

With Unit 3 operating at 88% power, the following symptoms occur:

- Reactor power INCREASING.
- Tave GREATER THAN Tref.
- Pressurizer Pressure INCREASING.
- Pressurizer Level INCREASING.

Which ONE of the following would cause the above symptoms to occur INITIALLY?

A. First Stage Turbine Pressure transmitter, PT-412A, Failed LOW.

B. Power range channel N-43 fails high.

C. Failed OPEN SG safety valve.

D. Uncontrolled rod withdrawal.

Proposed Answer:

D. Uncontrolled rod withdrawal.

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-AOPROD 3 (As available)

Question Source: Bank # INPO 20764
Modified Bank # (Note changes or attach parent)
New

Question History: 10/29/2001 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8,10
55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 20	Group #	2	
	K/A #	000028AK2.02	
OK	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and sensors and detectors

Proposed Question: Common 19

The plant is operating at 100% power with all control systems operating normally. The controlling channel, LT-460, reference leg of Pressurizer Level has just developed a leak where the reference leg connects to the D/P cell. Which one of the following best describes the immediate plant response from this leak?

A. LT-460 - indication will decrease, LT-459 indication will INCREASE, LT-461 - indication will INCREASE, charging flow will INCREASE.

B. LT-460 - indication will INCREASE, LT-459 indication will DECREASE, LT-461 - indication will DECREASE, charging flow will DECREASE.

C. LT-460 - indication will INCREASE, LT-459 indication will decrease, LT-461 indication will DECREASE, backup heaters will de-energize.

D. LT-460 - indication will DECREASE, LT-459 indication will DECREASE, LT-461 indication will DECREASE, backup heaters will energize.

Proposed Answer:

B. LT-460 - indication will INCREASE, LT-459 indication will DECREASE, LT-461 - indication will DECREASE, charging flow will DECREASE.

Explanation (Optional):

Technical Reference(s): SD-1.4 (Attach if not previously

________________ provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICPZLV E-5 (As available)

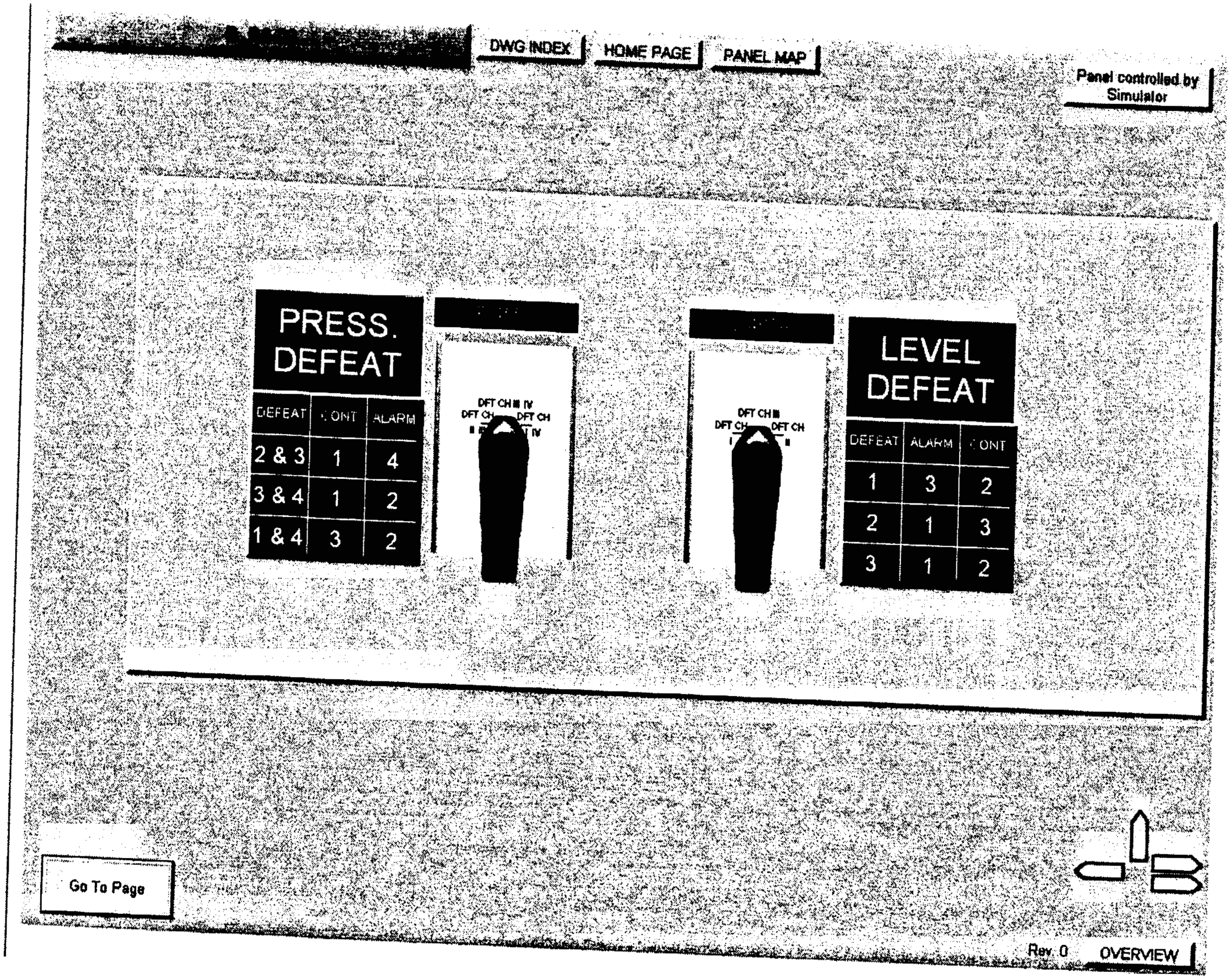
Question Source: Bank # INPO 24612
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: 5/30/2003 Seabrook 1

Question Cognitive Level: Memory or Fundamental Knowledge ________
Comprehension or Analysis X

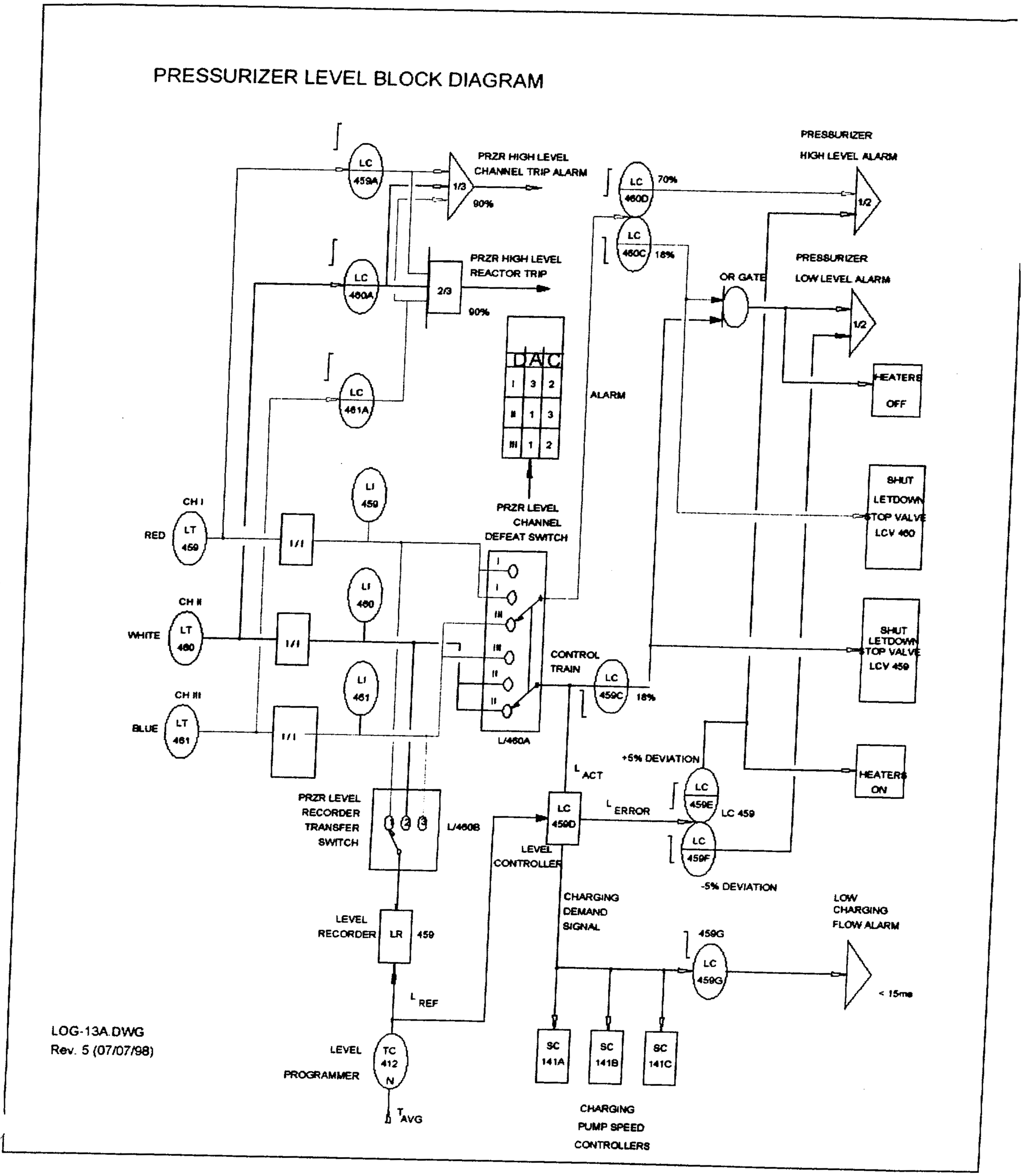
10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments:


DWG INDEX
HOME PAGE
PANEL MAP
Panel controlled by Simulator
PRESS. DEFEAT
DEFEAT CONT ALARM
2 & 3 1 4
3 & 4 1 2
1 & 4 3 2
DFT CH III IV
DFT CH II
DFT CH IV
DFT CH III
DFT CH I
DFT CH II
LEVEL DEFEAT
DEFEAT ALARM CONT
1 3 2
2 1 3
3 1 2
Go To Page
Rev. 0
OVERVIEW

ES #30

Figure 1.4-20: Pressurizer Level Block Diagram (LOG-13A)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 21	Group #	2	
	K/A #	000033AK1.01	
OK	Importance Rating	2.7	

Knowledge of the operational implications of the effects of voltage changes on performance as they apply to Loss of Intermediate Range Nuclear Instrumentation

Proposed Question: Common 20

Unit 3 is performing a plant shutdown. Power is 3% when alarm "Intermediate Range N36 Loss of Compensate Voltage," comes in.

How does this affect the Nuclear Instrumentation?

A. N36 reading would immediately drop about 1 decade. During the subsequent shutdown, SR NIs will energize automatically when N35 drops below P-6 setpoint.

B. N36 reading would immediately rise about 1 decade. During the subsequent shutdown, SR NIs will NOT energize automatically because N36 reading will remain above the P-6 setpoint.

C. N36 reading would NOT immediately change. During the subsequent shutdown, SR NIs will NOT energize automatically because N36 reading will remain above the P-6 setpoint.

D. N36 reading would NOT immediately change. During the subsequent shutdown, SR NIs will energize automatically when N35 drops below P-6 setpoint.

Proposed Answer:

C. N36 reading would NOT immediately change. During the subsequent shutdown, SR NIs will NOT energize automatically because N36 reading will remain above the P-6 setpoint.

Explanation (Optional):

Technical Reference(s): 3-AOP-NI-1
SD-13
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICEXC E-3 (As available)

Question Source: Bank # INPO 26088
Modified Bank # (Note changes or attach parent)
New

Question History: 9/1/2003 Prairie Island 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8,10
55.43

Comments:

WS #21

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.13 __Has a failure in Intermediate Range NIs occurred?	__ **GO TO** Step 4.19.
4.14 __Is remaining Intermediate Range channel inoperable?	__ **GO TO** Step 4.16.
4.15 __Do at least 2 of 4 Power Range channels indicate > 10%?	__ **IF NOT** already being met, **THEN** meet requirements of TS 3.3.1.

<u>CAUTION</u>

Failure of either Intermediate Range channel in the high direction will block automatic re-energizing of Source Range High Voltage. Simultaneous operation of both Intermediate Range Permissive Defeat Pushbuttons (when remaining Intermediate Range channel indicates < 3.6E-11) will be required to obtain Source Range operation.

4.16 __**INITIATE** applicable section of SOP-NI-001 (Excore Nuclear Instrumentation System Operation) to remove affected channel from service.	
4.17 Notify the following of status of NI failures: __Site Operations Manager __Unit Operations Manager __Reactor Engineer	
4.18 __**RETURN** to procedure and step in effect.	

END

the inner and outer cylinders), the neutrons react with the boron causing ionization.

$$_5B^{10} + {_0N^1} \rightarrow (_5B^{11})^* \rightarrow {_3Li^{7+++}} + {_2\alpha^{4++}} + \text{energy}$$

The lithium and alpha particle resulting from this reaction cause secondary ionization in the outer can. The electrons produced by the ionization are collected on the outer can wall. This produces a signal that is proportional to the neutron flux. Electrons are also collected on the outer can wall from the gamma radiation, which interacts with the outer gas volume. This additional signal is proportional to the gamma flux and is additive to the neutron flux signal. The outer can operates in the Ionization Region; thus, all the charged particles produced in the initial ionizing events are collected on the electrodes.

In the inner can, the gamma flux also reacts with the N_2 gas, producing a signal proportional to the gamma radiation. The inner can is operated in the Recombination Region to permit adjustment of the output current by varying its applied voltage. The inner can voltage is called the compensating voltage. If the compensation voltage is set properly, the outer can signal due to gamma plus neutron flux, will interact with the inner can gamma flux only signal. The gamma signals cancel out leaving the neutron only signal which is then amplified before it is displayed on the meter or sent to the protection and control circuitry.

2.10.2 Gamma Compensation in the Intermediate Range

It is necessary to define the term compensation and the effects of "under-compensation" and "over-compensation" to clearly understand the process of neutron detection in the intermediate range.

Compensation is a term applied to the negative voltage signal applied to the inner can of the CIC which cancels or compensates for the current signal produced by the gamma radiation interacting within the outer can of the detector. This becomes very important to the operator because an incorrect setting of compensating voltage, i.e. over-compensation or under-compensation, would cause an erroneous neutron level indication on the meters, as shown on Figure 13-22.

- Over-compensation occurs when the compensation voltage is set to high. This results in a higher current due to gamma flux in the inner can than is being generated in the outer can due to the same gamma flux. The results of this mismatch is that part of the current due to the neutron flux is also cancelled, causing the indicated current level to be less than actual.

- Under-compensation occurs when the compensation voltage is less than that required. The current due to gamma from the inner can is now smaller than the current due to gamma from the outer can. This allows some current due to gamma in the outer can to remain and add to the neutron flux resulting in increased output current from the detector causing it to indicate above actual levels.

To obtain this true neutron-only signal, the two opposing gamma signals must be cancelled exactly. Since it is physically impossible for both the inner and outer cans to be manufactured identically sensitive to the gamma flux present under all operating conditions, the problem of how to ensure exact compensation arises. By grooving the inner electrode and applying a variable negative voltage, the size of the inner can is adjusted electrically. The inner can of the CIC operates in the recombination region of the detector characteristic curve and, by adjusting the compensating voltage, only a fraction of the total ionization is collected.

The IR drawer monitors reactor power over a range of eight decades between 10^{-11} and 10^{-3} ion chamber amperes. Indications of level and startup rate (SUR) are provided at the NIS cabinets, and on panel FCF.

Because neutron events are occurring at a high rate, no signal conditioning is necessary prior to the log current amplifiers. A block diagram of the intermediate range in provided as Figure 13-23.

2.10.3 Log Current Amplifier

This assembly receives current from the detector in the range between 10^{-11} and 10^{-3} amperes. The assembly provides a logarithmic voltage output, 0 to 10 VDC, proportional to a linear input current. With the use of the log amplifier, the wide range current input is compressed logarithmically to a usable voltage suitable for metering and the generation of trip signals. (Figure 13-23 provides a block diagram of the intermediate range.) The output from the log amplifier is simultaneously coupled to an isolation amplifier and four bistable relay drivers. The output is also displayed on the neutron level meter calibrated in amperes between 10^{-11} and 10^{-3}.amps.

Internal switches and potentiometers are provided for setting and adjusting the log current amplifiers. Both fixed and variable signals can be injected into the log amplifiers for testing and calibration purposes. This is accomplished by the use of switches located on the front panel of the drawer and a calibrate module located inside of the IR drawer assembly.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 22	Group #	2	
	K/A #	000037AA2.12	
OK	Importance Rating	3.3	

Ability to determine and interpret the flow rate of leak as it applies to the Steam Generator Tube Leak

Proposed Question: Common 21

A SG tube leak is in progress. Plant conditions just before the leak were steady state with no evolutions in progress. Some time later, the following conditions exist:

- CVCS charging flow rate = 63 gpm
- CVCS letdown flow = 75 gpm
- Total RCP seal injection = 32 gpm
- Total RCP seal leakoff flow = 12 gpm
- RCS temperature at no load Tave and steady
- PZR Press and Level are stable

Based on the above indications, what is the approximate RCS SG leak rate?

A. 1 gpm

B. 8 gpm

C. 20 gpm

D. 28 gpm

Proposed Answer:

B. 8 gpm

Explanation (Optional):

Technical Reference(s): SD-3.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-AOPSG1 A (As available)

Question Source: Bank # INPO 20216
Modified Bank # (Note changes or attach parent)
New

Question History: 9/10/2001 Cook 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 23	Group #	2	
	K/A #	000051G2.2.12	
OK	Importance Rating	3.0	

Knowledge of the surveillance procedures

Proposed Question: Common 22

Unit 3 is running 3-PT-V089, Online Turbine Mechanical Trip Features Test.

The "Test Handle" has just been placed in the TEST position in preparation for doing the Low Vacuum Trip Test (Simulated).

The control room reports that actual condenser vacuum has dropped to the turbine trip setpoint.

With no operator action, which one of the following will occur?

A. The turbine low vacuum trip device will NOT actuate, and the turbine will NOT trip.

B. The turbine low vacuum trip device will actuate, and the turbine will trip.

C. The turbine low vacuum trip device will NOT actuate, but the turbine will trip.

D. The turbine low vacuum trip device will actuate, but the turbine will NOT trip.

Proposed Answer:

D. The turbine low vacuum trip device will actuate, but the turbine will NOT trip.

Explanation (Optional):

Technical Reference(s): I3LP-ILO-MTG001 Page 30,31 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-MTG001 4 (1583) (As available)

Question Source: Bank # INPO 22518
Modified Bank # (Note changes or attach parent)
New

Question History: 10/1/2002 Diablo Canyon 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4,10
55.43

Comments:

LOS #23

4. Hydraulic induced trip dumps high pressure autostop oil pressure off of the overspeed trip valves by operation of a pivot plate arrangement

   a. 20 AST solenoid trip

   b. Low bearing oil pressure trip

   c. Low vacuum trip

   d. Thrust bearing wear trip

   e. In addition, a manual trip from the front standard and the mechanical overspeed trip device also cause a hydraulic trip through the overspeed trip device

   f. Once HP autostop oil dumps, low pressure autostop oil dumps and the trip process continues the same as for a solenoid trip

5. Autostop oil header is monitored by three pressure switches (63/AST 2, 3, 4). They are arranged into three redundant 2/3 logic circuits — Logic Diagram 565D172 Sht 2

   a. Two 2/3 logics are used to trip the reactor above P-8 ($\geq$35% Rx Pwr)

   b. One 2/3 logic trips the generator upon initiation of a turbine trip

   c. If 1/3 pressure switches senses AST oil pressure <45 psig, alarms at panel FAF (normal pressure 120 psig)

6. Low bearing oil pressure trip — TP-14

   a. Prevents overheating and possible bearing damage

   b. Pressure of 6.5 psig at No. 1 bearing initiates trip

   c. Pressure switch (63/LBO) in the bearing supply actuates the low bearing pressure alarm on panel FAF

7. Low vacuum trip — E.O.4

a. Prevents excessive heating of low pressure turbine exhaust hood and last row of turbine blades

b. Turbine is tripped if condenser vacuum falls below 18" mercury

c. Trip feature may be bypassed locally during startup and automatically engages when vacuum reaches >18 inches mercury or >3psig back pressure

d. Pressure switch 63/LV provides first out alarm at panel FAF

e. A special drip leg (trap) on the vacuum sensing line keeps any oil leakage from being drawn into the main condenser

8. Thrust bearing wear trip

a. It protects the turbine from mechanical damage when excessive thrust bearing wear on shoes or excessive axial movement is detected

b. One detector

(1) Located on governor end and senses thrust in both directions using a back press signal

(2) Setpoint 43 psig, alarms on panel FAF as sensed by PS-63/TB

9. Manual Trip

a. Lever located on front standard to manually dump autostop oil

b. Button on CR flight panel opens both solenoid valves to dump autostop oil

c. Either manual trip actuates the first out MANUAL TURBINE TRIP alarm at panel FAF

10. Generator Trip

a. Prevents excessive overspeed conditions

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 24	Group #	2	
	K/A #	000068AA2.06	
OK	Importance Rating	4.1	

Ability to determine and interpret RCS pressure as it applies to the Control Room Evacuation

Proposed Question: Common 23

The control room has been evacuated due to a fire with heavy smoke. Offsite power has been lost. All three EDGs are supplying their respective buses. Which of the following describes the primary method that will be used to control RCS pressure?

A. Manually opening and closing one Pressurizer PORV.

B. Adjusting 31 Charging Pump speed to control level and thus pressure.

C. Energizing and de-energizing 31 Pressurizer backup heaters.

D. Local operation of Auxiliary Spray.

Proposed Answer:
C. Energizing and de-energizing 31 Pressurizer backup heaters.

Explanation (Optional):

Technical Reference(s): 3-AOP-SSD-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # ____________
Modified Bank # ____________ (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____________

10 CFR Part 55 Content: 55.41 5,10
55.43 5

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.27 __ Are any Containment Spray Pump(s) running?	__ **GO TO** Step 4.29.
4.28 __ Remove control power fuses and open 480V breakers for running Containment Spray Pumps.	1.__ Secure associated 480V Bus(es) by opening bus supply breaker or shutting down associated EDG. 2. **IF** pump(s) are still running, **THEN** dispatch NPO to close the following valves: (PAB 68ft., Piping Pen Area): __ SI-869A (31 SPRAY PUMP DISCHARGE LINE ISOLATION) __ SI-869B (32 SPRAY PUMP DISCHARGE LINE ISOLATION)
4.29 __ Are any RHR pumps running that were not previously running?	__ **GO TO** Step 4.31.
4.30 __ Remove control power fuses and open 480V breaker for running RHR Pumps.	__ Secure associated 480V Bus(es) by opening bus supply breaker or shutting down associated EDG.
4.31 __ Are any HHSI pumps running?	__ **GO TO** Step 4.34.
4.32 __ Is PRZR level > 5% and subcooling > 40°F?	__ **GO TO** Step 4.34.
4.33 __ Remove control power fuses and open 480V breaker for running HHSI Pumps.	__ Secure associated 480V Bus(es) by opening bus supply breaker or shutting down associated EDG.
4.34 __ Are all PRZR heaters de-energized?	__ **IF** more than one PRZR heater is energized, **THEN** remove control power fuses and open breakers to all but one heater (31 heater preferred - controlled by CRS).
4.35 __ Are two or more SW Pumps running on essential header? (CB, 15' el.)	__ **PERFORM** Attachment 11 (Service Water Pump Runout Protection During Pump Start) (Page 135).
4.36 Are at least two of the following 480V safeguards buses energized? __ 2A/3A __ 5A __ 6A	__ **INITIATE** applicable sections of SOP-EL-1 (Diesel Generator Operation) to attempt power restoration to at least two safeguards buses.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 25	Group #	2	
	K/A #	W/E14AK3.01	
OK	Importance Rating	3.8	

Knowledge of the reasons for guidance contained in EOP for loss of containment integrity

Proposed Question: Common 24

Given the following plant conditions:

- The Unit has experienced a fault on 31 Steam Generator inside containment.
- The crew has transitioned from E-0 to E-2, Faulted Steam Generator Isolation.
- Containment pressure is currently at 28 psig and slowly rising.
- Both Containment Spray Pumps are NOT operating.

Which ONE of the following indicates the correct action for the crew to take?

A. Continue in E-2, Faulted Steam Generator Isolation and transition to FR-Z.1, Response To High Containment Pressure if containment pressure exceeds 46 psig.

B. Continue in E-2, then use ES-0.0, Rediagnosis (if necessary) to transition to the correct procedure

C. Immediately transition to FR-Z.1, Response To High Containment Pressure.

D. Go to E-0, Reactor Trip or Safety Injection and revalidate SI automatic actions to ensure Containment Fan Cooler Units are operating properly.

Proposed Answer:

C. Immediately transition to FR-Z.1, Response To High Containment Pressure.

Explanation (Optional):

Technical Reference(s): F-0.5 Containment Status Tree (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRZ 8 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

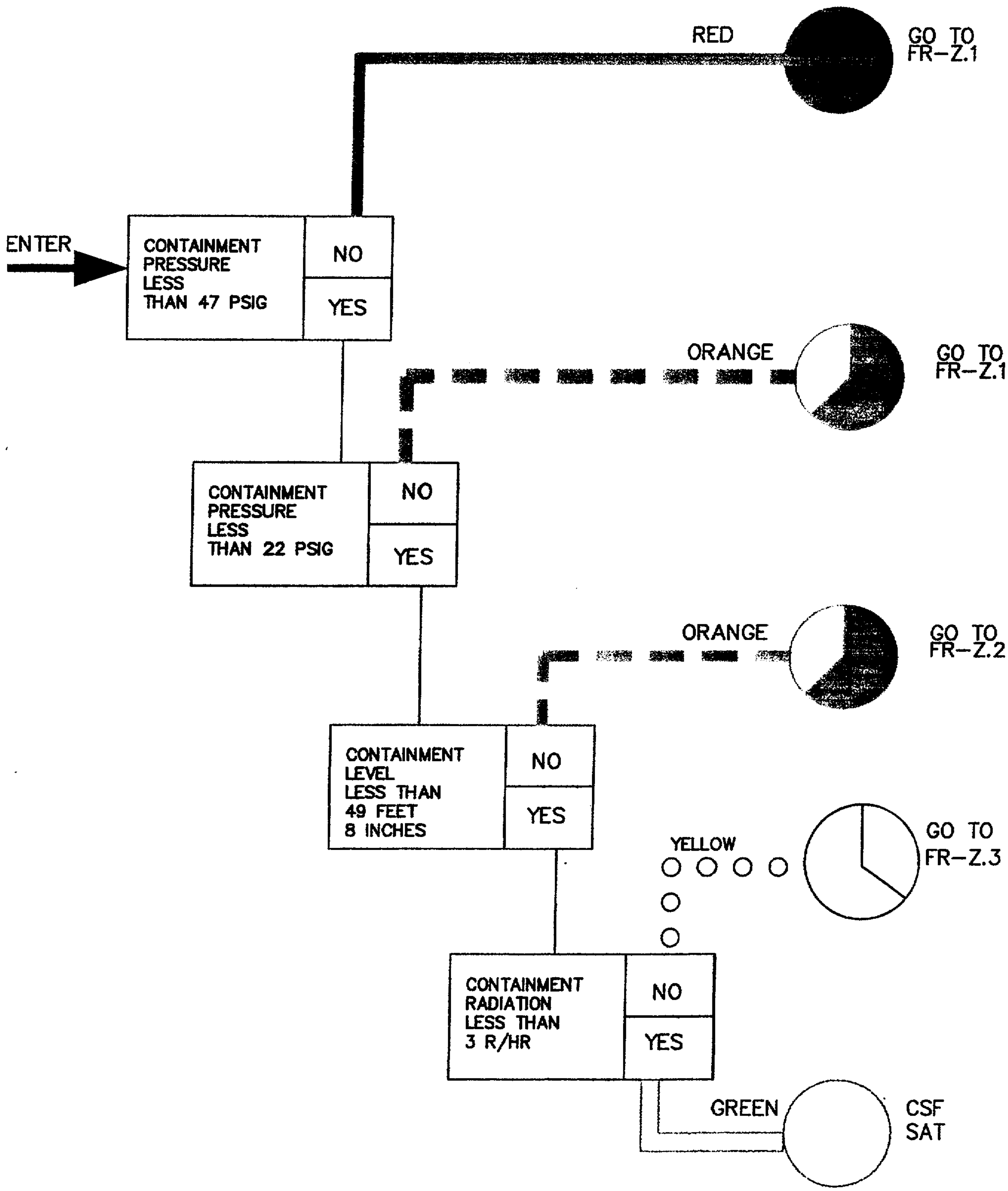
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5,10
55.43 5

Comments:

WS #25

Number:	Title:	Revision
F-0.5	CONTAINMENT	8



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 26	Group #	2	
	K/A #	000076AA1.04	
OK	Importance Rating	3.2	

Ability to operate and / or monitor the failed fuel-monitoring equipment as they apply to the High Reactor Coolant Activity

Proposed Question: Common 25

In accordance with the abnormal operating procedure 3-AOP-HIACT-1, High Activity, what should the operators do once Chemistry verifies the high activity condition?

A. Place Excess letdown in service in addition to normal letdown.

B. Divert letdown to CVCS HUT and maximize makeup.

C. Remove cation demineralizer from service.

D. Maximize letdown flow.

Proposed Answer:
D. Maximize letdown flow.

Explanation (Optional):

Technical Reference(s): 3-AOP-HIACT-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-AOPACT 3 (As available)

Question Source: Bank # INPO 24663
Modified Bank # (Note changes or attach parent)
New

Question History: 5/30/2003 Seabrook 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS #26

4. SUBSEQUENT ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- Large or sudden changes in RCS temperature or flow could cause a crud burst.
- If RCS activity is ≥ ***TO 25% OF*** TS 3.4.16 limits, the plant is required to be in MODE 3 with T_{ave} < 500°F within 6 hours.

TPC 05-00326

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.1 __ **IAAT** RCS activity is ≥ ***TO 25% OF*** TS 3.4.16 limits **AND** reactor is in MODE 1 or MODE 2, **THEN INITIATE** plant shutdown according to applicable POP.	
4.2 __ **IAAT** <u>all</u> the following conditions exist: __ RCS activity ≥ ***TO 25% OF*** TS 3.4.16 limits __ Reactor is in MODE 3 __ T_{ave} ≥ 500°F **THEN INITIATE** POP-3.3 (Plant Cooldown – Hot To Cold Shutdown) to reduce T_{ave} < 500°F.	

NOTE

- If activity on R-63 is > 5 μCi/cc but < 50 μCi/cc, then HP should assist Chemistry in primary sampling.
- If activity on R-63 is > 50 μCi/cc, then consideration should be given to using the Post-Accident Sampling System.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.3 __ Notify Chemistry to periodically sample for RCS activity <u>and</u> trend results.	
4.4 __ Is letdown aligned through CVCS demineralizers?	__ **PERFORM** applicable section of SOP-CVCS-004 (Placing the CVCS Demineralizers In or Out of Service) to place mixed bed demin in service.
4.5 __ **INITIATE** applicable section of SOP-CVCS-002 (Charging, Seal Water, and Letdown Control) to increase letdown flow up to a maximum of 120 gpm.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 27	Group #	2	
	K/A #	W/E08G2.1.23	
OK	Importance Rating	3.9	

Ability to perform specific system and integrated plant procedures during all modes of plant operation

Proposed Question: Common 26

On the current outage schedule, the RCS cooldown (from 340°F to <200°F) is supposed to occur in the next 3 hours. You have just finished placing RHR in service for cooldown but during this evolution you determined that 31 RHR HX had a tube leak. 31 RHR HX is currently isolated. How will this impact the RCS cooldown on the outage schedule?

A. The cooldown will be completed on schedule because the RHR system has two 100% redundant trains.

B. The RCS cooldown will be completed but it will take longer but not more than twice as long as scheduled.

C. The cooldown will be completed on schedule because the SGs will do most of the cooling until RCS temperature is below 212°F.

D. The RCS cooldown can not be completed until decay heat level drops below the capacity of the single RHR train.

Proposed Answer:

B. The RCS cooldown will be completed but it will take longer but not more than twice as long as scheduled.

Explanation (Optional):

Technical Reference(s): ____________ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RHR001 6 (As available)

Question Source: Bank # INPO 26098
Modified Bank # ____________ (Note changes or attach parent)
New ____________

Question History: *9/1/2003* *Prairie Island 1*

Question Cognitive Level: Memory or Fundamental Knowledge ____________
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 *10*
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 28	Group #	1	
	K/A #	003A4.07	
OK	Importance Rating	2.6	

Ability to manually operate and / or monitor in the control room RCP seal bypass

Proposed Question: Common 27

Given the following conditions:

- Plant cooldown is in progress.
- RCS temperature is 220°F.
- RCS pressure is 375 psig.
- VCT pressure is 25 psig.
- RCPs are operating with all No.1 Seal Leakoff valves open.
- All RCP seal injection flows are 8 gpm
- RCP seal discharge valves 261A-D are open
- 32 RCP #1 seal leakoff flow indicates 0.8 gpm and slowly decreasing.
- 32 RCP lower radial bearing temperature is 195°F and slowly rising

Which ONE (1) of the following actions is required?

A. OPEN RCP Seal Bypass Valve, 246, to increase seal leakoff flow.

B. CLOSE HCV-142, Charging Line Flow Control Valve, to increase seal injection flow.

C. Trip operating RCPs and isolate seal leakoff due to insufficient seal DP.

D. Isolate #1 seal leakoff for 32 RCP to increase #1 seal DP.

Proposed Answer:

A. OPEN RCP Seal Bypass Valve, 246, to increase seal leakoff flow.

Explanation (Optional):

Technical Reference(s): 3-SOP-RCS-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSRCP B (As available)

Question Source: Bank # INPO 28048
Modified Bank # (Note changes or attach parent)
New

Question History: 9/27/2004 Robinson 2

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

LOS #28

____ 4.3.3 IF securing 34 RCP, THEN PLACE RC-PCV-455A, Pressurizer Spray Pressure Control Valve, in MANUAL and CLOSE.

____ 4.3.4 MAINTAIN CCW flow to motor bearing oil coolers for at least 30 minutes after shutdown.

____ 4.3.5 MAINTAIN CCW flow to RCP thermal barrier until RCS temp is less than 150°F OR MAINTAIN seal injection between 6-12 gpm.

____ 4.3.6 MAINTAIN RCP seal injection flow until RCS pressure is less than 100 psig.

____ 4.3.7 LOG RCP shutdown in Unit Log.

4.4 Operation of RCP Seal Bypass Valve

NOTE

Bypass line allows additional seal injection flow through pump bearing for cooling.

____ 4.4.1 IF RCP lower radial bearing temp, as monitored via RCP seal inlet temp indicator TI-155, TI-154, TI-153, or TI-152 exceeds 190°F (Ref. 5.2.12) OR No. 1 seal outlet temp indicator TI-148, TI-146, TI-132, or TI-125 exceeds 200°F, THEN:

____ 4.4.1.1 VERIFY all of the following conditions are met:

- ____ Continued operation of applicable RCP is required.
- ____ RCS pressure is greater than 100 psig and less than 1000 psig.
- ____ RCP seal discharge valves are OPEN:
  - ____ 31 VLV 261A
  - ____ 32 VLV 261B
  - ____ 33 VLV 261C
  - ____ 34 VLV 261D
- ____ RCP seal return flow as indicated on FR-158, FR-159, FR-156, or FR-157 is less than 1 gpm (Ref. 5.2.12).
- ____ Seal injection flow rate to each pump is 6 to 12 gpm.

____ 4.4.1.2 OPEN RCP Seal Bypass Valve No. 246.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 30	Group #	1	
	K/A #	004K4.01	
OK	Importance Rating	2.8	

Knowledge of CVCS design feature and / or interlock which provide for oxygen control in the RCS

Proposed Question: Common 28

Hydrogen is supplied to the Volume Control Tank (VCT) via an automatic pressure regulator.

This design feature of the CVCS system is provided to ________________.

A. lower iodine levels in the RCS.

B. minimize oxygen in the RCS.

C. control the pH in the RCS.

D. maintain corrosion product solubility in the RCS.

Proposed Answer:
B. minimize oxygen in the RCS.

Explanation (Optional):

Technical Reference(s): SD 3.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CVC001 3.a. (As available)

Question Source: Bank # INPO 21571
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: 7/17/2002 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

downstream of TCV-149 and the demineralizers in order to ensure that whichever flowpath the letdown stream takes it must pass through the filter. Figure 3.0-1 shows the location of the filter. Physically, the reactor coolant filter is located in the Primary Auxiliary Building at the 73 foot elevation. Flow enters the filter near the top, passes down through filter cartridges and out near the bottom. The filter can be bypassed when needed for filter cartridge replacement. The filter cartridges are replaced when the pressure drop across them reaches approximately 20 psid or when the filter housing reads 5 to 10 Rem/hr on contact or in accordance with applicable RES department procedures. Vent and drain connections are provided for use in filter cartridge replacement.

2.4 Volume Control Tank (Figure 3.0-17)

Letdown flow continues to the VCT level control valve (LCV-112A) which directs the flow to the VCT or to the CVCS hold up tanks (Hut's). LCV-112A is normally aligned to the VCT, however, it automatically shifts to the Hut's on a high VCT level of 92% (reset 83%). Level control is discussed in detail in section 2.4.1. A normal-divert switch on the CVCS supervisory panel SFF in the control room allows the operator to manually divert flow to the HUT regardless of VCT level. Upon loss of air pressure or electrical power, LCV-112A fails to the VCT position.

The VCT provides an additional surge capacity for the reactor coolant system that is not accommodated by the pressurizer following load transients. It provides a convenient point for adding hydrogen to the coolant for RCS Oxygen control during operation and a means of degassing the RCS during shutdown and cooldown. The VCT also provides a suction head for the charging pumps and the backpressure for the Reactor Coolant Pump (RCP) seals. The VCT has a capacity of 3000 gallons and is located in the Primary Auxiliary Building at the 73 foot elevation.

The letdown flow enters the VCT through a spray nozzle located at the top of the tank. The spray nozzle enhances coolant hydrogen absorption by increasing the water to gas blanket contact area. The vapor space is predominantly hydrogen. It also provides a scrubbing action for the removal of fission product and other non-condensable gases. A hydrogen gas blanket is maintained on the VCT to control the RCS hydrogen concentration between 25 and 35 cc/kg. This ensures that sufficient hydrogen concentration is available for reaction with oxygen and subsequently assists with corrosion control. Adjustments in the hydrogen concentration are made by changing the setpoint of

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 31	Group #	1	
	K/A #	005K5.09	
OK	Importance Rating	3.2	

Knowledge of the operational implications of dilution and boration considerations

Proposed Question: Common 29

Given the following conditions:

The plant is being cooled down to 140°F for maintenance which will NOT require the RCS be opened. The crew is in the process of placing the first Residual Heat Removal (RHR) train in service for RCS cooling. Current RCS temperature is 345°F.

Current boron concentrations are as follows:

- RHR (train to be placed in service) boron 1020 ppm
- Required Shutdown Margin at 300°F boron 1750 ppm
- Required Shutdown Margin at 68°F boron 1800 ppm
- RCS boron 2025 ppm
- Refueling boron 2050 ppm

Before the RHR train can be placed in service for RCS cooling, RHR boron concentration must be increased by a MINIMUM of ...

A. 730

B. 780

C. 1005

E. 1030

Proposed Answer:

B. 780

Explanation (Optional):

Technical Reference(s): Graph RCS-4A
3-SOP-RHR-001
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RHR001 8.a. (1170) (As available)

Question Source: Bank # INPO 27468
Modified Bank # 27468 (Note changes or attach parent)
New

Question History: 3/24/2004 Harris 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

1.0 PURPOSE

1.1 This procedure establishes requirements for RHR System operation.

1.2 This procedure applies to the RHR System.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 WHEN RCS pressure is greater than OR equal to 400 psig OR RCS temp is greater than OR equal to 350°F, THEN RHR SHALL NOT be in service (Ref 5.2.8).

2.2 IF RCS pressure increases to greater than or equal to 550 psig, THEN AC-MOV-730, RHR Loop Suction Isolation, and AC-MOV-731, RHR Loop Suction Isolation, will automatically close: (Ref 5.2.9)

- IF AC-MOV-730 OR AC-MOV-731 close, THEN RCS pressure must be lowered to less than 450 psig to enable valve reopening.
- WHEN AC-MOV-730 OR AC-MOV-731 start to close, THEN valves can NOT be reopened until fully closed.

2.3 RCS and RHR boron concentrations SHALL be greater than or equal to minimum required concentration for shutdown per Graphs RCS-4A and RCS-4B, Minimum Required Boron for Shutdown.

2.4 WHEN RHR warmup is complete, THEN RHR flow SHALL be maintained as follows:

- Total RHR flow (recirc + miniflow + core flow) greater than 1170 gpm to ensure minimum flow for continuous RHR pump operation **{Ref. 5.1.1}**.
- Core flow greater than 1000 gpm to ensure adequate RCS mixing

2.5 To prevent equipment damage, flow through a single RHR pump:(Ref 5.2.10)

- with 1 RHR HX in service flow SHALL NOT exceed 3000 gpm.
- with 1 RHR Pump in service flow SHALL NOT exceed 4500 gpm.

CAUTION

- WHEN RHR pump flow is greater than 300 gpm AND less than 1170 gpm THEN RHR pump operation should be limited to 3 hrs or less in a 24-hr period (Ref 5.2.11)
- RHR pump minimum flow SHALL be maintained greater than or equal to 100 gpm during start/stop pump operation, which is consider to be 30 min or less (Ref 5.2.11)

____ 4.2.2.4 START desired RHR pump.

____ 4.2.2.5 WAIT at least 10 min (Ref 5.2.16) to allow for proper mixing.

____ 4.2.2.6 REQUEST Chemistry to sample boron concentration.

____ 4.2.2.7 WHEN sample has been obtained, THEN STOP desired RHR pump.

____ 4.2.2.8 IF sample results indicate RHR loop boron concentration is equal to or greater than the 68°F (Ref 5.2.17) Cold Shutdown boron concentration, THEN GO TO Step 4.2.3.

____ 4.2.2.9 IF sample results indicate RHR loop boron concentration is less than RCS boron concentration AND RX Trip breakers are open, THEN:

____ a) REVIEW Unit Log to ensure motor starting requirements of 3-SOP-EL-004A, Electric Motor Operation, will be met.

____ b) START RHR Pump.

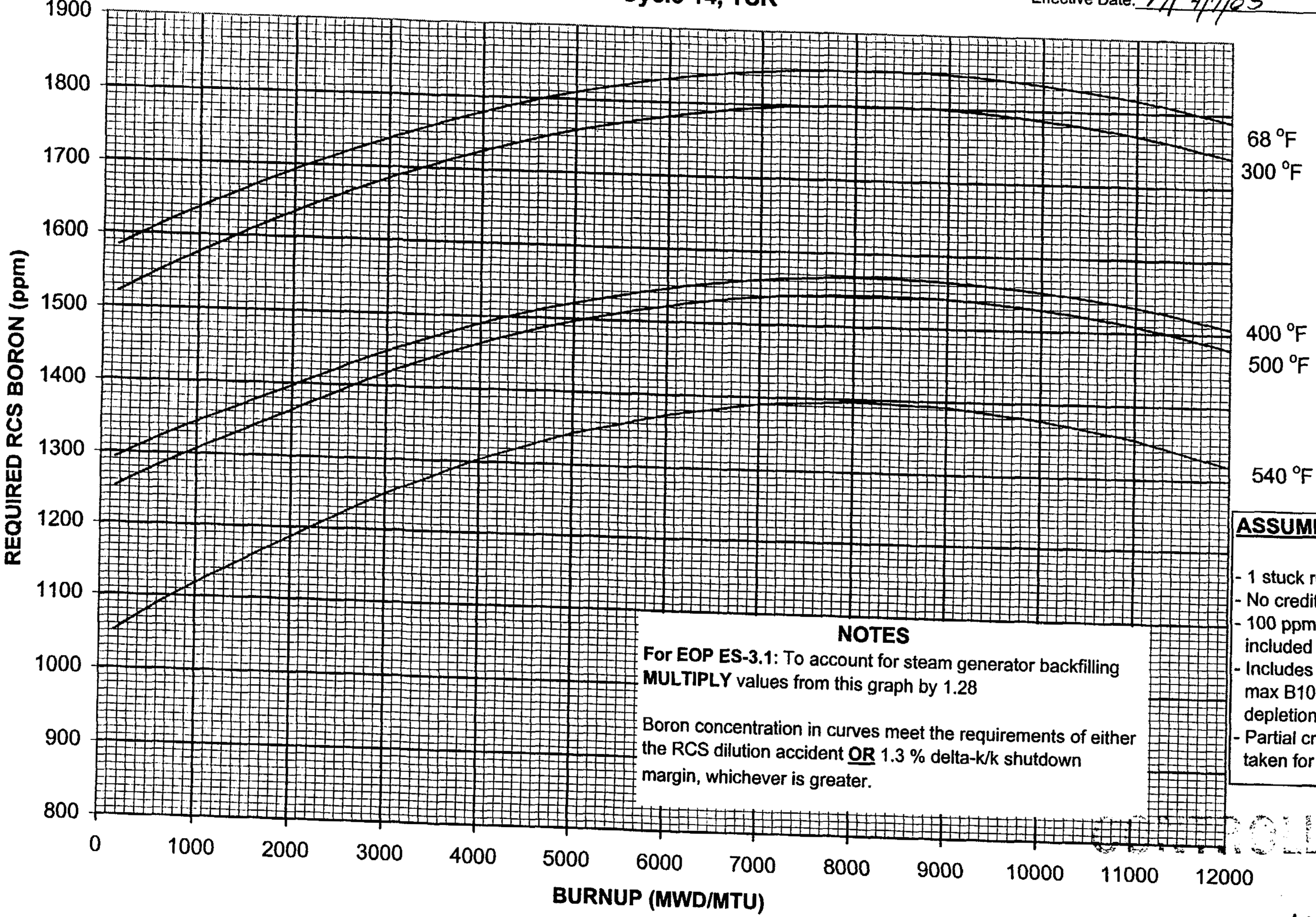
____ c) ENSURE letdown pressure as indicated on PI-135, LP Letdown Press, is less than RHR pump discharge pressure as indicated locally on PI-635, RHR Pump Discharge Pressure Indicator, by adjusting PCV-135, Low Pressure Letdown Line Backpressure Control Valve.

____ d) ENSURE a CVCS HUT is aligned to receive CVCS letdown per 3-SOP-CVCS-001, CVCS Holdup Tank Operation.

____ e) ENSURE CH-LCV-112A, VCT Inlet Diversion, is in DIVERT.

RCS-4A Rev. 20
Minimum Required Boron for Shutdown
Cycle 14, TSR

Written By: [signature]
Reviewed By: [signature]
Approved By: [signature]
Effective Date: 4/7/05



NOTES

For EOP ES-3.1: To account for steam generator backfilling **MULTIPLY** values from this graph by 1.28

Boron concentration in curves meet the requirements of either the RCS dilution accident **OR** 1.3 % delta-k/k shutdown margin, whichever is greater.

ASSUMPTIONS

- 1 stuck rod
- No credit for Xe
- 100 ppm SF included
- Includes effect of max B10 depletion
- Partial credit taken for Sm

CONTROLLED

COPY # 424

WS #3

QuestionId	27468	ExamType	ILO	ExamDate	3/24/2004
AbbrevLocName	Harris 1	NSSSVendor	#Name?	NSSSType	PWR

QuestionStem

Given the following conditions:

The plant is being cooled down to 140oF for maintenance which will NOT require the RCS be opened.
The crew is in the process of placing the first Residual Heat Removal (RHR) train in service for RCS cc
Current boron concentrations are as follows:

RHR (train to be placed in service) boron1021 ppm
Required Shutdown Margin boron1200 ppm
RCS boron1341 ppm
Cold Shutdown boron1750 ppm
Refueling boron2261 ppm

Before the RHR train can be placed in service for RCS cooling, RHR boron concentration must be incr
MINIMUM of

QuestionCommen

CognitiveLevel		ExamLevel	R	RefMaterial		ParentQuestionId	

Answer: 179 ppm.

Distract1: 320 ppm. Distract1Co

Distract2: 729 ppm. Distract2Co

Distract3: 1240 ppm. Distract3Co

KaNumber	005K5.09 67
KaSegment1	
KaSegment2	
KaSegment3	NULL
KaSegment4	NU
KaSegment5	N
KaRevision	

ooling.

eased by a

Reference Req'd (Y/N)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 32	Group #	1	
	K/A #	006A1.18	
OK	Importance Rating	4.0	

Ability to predict and / or monitor changes in parameters to prevent exceeding design limits associated with operating the ECCS controls including PZR level and pressure

Proposed Question: Common 30

Given the following conditions:

- A LOCA has occurred.
- The crew is performing actions of ES-1.2, Post LOCA Cooldown And Depressurization.
- Pressurizer level is stable at 58%.
- RCS pressure is stable at 1280 psig.
- The CRS determines that a SI pump can be stopped in accordance with ES-1.2.

When the RO stops the SI Pump, which one of the following describes the Pressurizer level response?

A. PRZR level will remain at its current value.

B. PRZR level will rise until charging is realigned to the VCT.

C. PRZR level will drop until normal charging and letdown are restored

D. PRZR level will drop until RCS pressure stabilizes at a lower value, then will stabilize.

Proposed Answer:

D. PRZR level will drop until RCS pressure stabilizes at a lower value, then will stabilize.

Explanation (Optional):

Technical Reference(s): ES-1.2, Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPE10 6 (As available)

Question Source: Bank # INPO 24940
Modified Bank # (Note changes or attach parent)
New

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

As noted in Table 2, the subcooling criteria for stopping the high-head SI pumps are given in two ways. For the case "w/o throttling," maximum charging flow from the two charging pumps is assumed at all times, before and after the SI flow reduction. For the case "with throttling," the charging flow is adjusted and the following sequence of actions is performed when one high-head SI pump is stopped:

1) Reduce charging flow to a minimum (zero) while maintaining at least a minimum subcooling (errors).
2) Allow RCS subcooling to increase to the required criterion for stopping the next SI pump. The subcooling will increase as the RCS cooldown continues.
3) Stop SI pump and immediately increase charging flow to a maximum.

By throttling the charging flow in this manner, the injection flow reduction when stopping an SI pump will be smaller than the flow reduction for the "w/o throttling" case. The subcooling criterion is also reduced when throttling is used.

In the transient analysis presented here, the first (and second) high-head SI pump was stopped without throttling the charging flow, consistent with the way the ES-1.2 guideline is structured. Since charging flow is a small percentage of the total injection flow, the subcooling criteria with and without throttling are not significantly different, 55°F versus 45°F for stopping the first high-head SI pump.

At t= 60:00, RCS subcooling had increased to 55°F and PRZR level was 46% and increasing. The first high-head SI pump was then stopped (Step 11). Over the next five minutes, RCS pressure decreased (from 1220 psig) and stabilized at 1100 psig. PRZR level decreased to 40% and continued to decrease slowly as the cooldown continued.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 33	Group #	1	
	K/A #	007G2.1.32	
OK	Importance Rating	3.4	

Ability to explain and apply all system limits and precautions

Proposed Question: Common 31

Which of the following describes the adverse affects of NOT maintaining the Pressurizer Relief Tank (PRT) within design level band?

A. If the level is too high, the tank will overflow to CNMT sump causing possible false indications of RCS leakage to CNMT.

B. If the level is too high, the sparger pipe will be too far underwater rendering the cooling affect of makeup water ineffective.

C. If the level is too low, there would be insufficient water volume to absorb and condense a design discharge of PRZR safety leading to possible over temperature and overpressure of the PRT.

D. If the level is too low the radioactive gases that leak from the top of the PRZR would not be adequately scrubbed, thus causing subsequent elevated gaseous activity levels inside CNMT.

Proposed Answer:

C. If the level is too low, there would be insufficient water volume to absorb and condense a design discharge of PRZR safety leading to possible over temperature and overpressure of the PRT.

Explanation (Optional):

Technical Reference(s): 3-SOP-RCS-007 (Attach if not previously

SD-1.4 provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSPZR E-8 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5

Comments:

WS #33

2.3.5 Acoustical Monitoring of Code Safeties and PORVs

The safety and relief valve acoustical monitor system provides control room indications representative of valve position. The parameter actually monitored is flow.

One piezoelectric accelerometer is clamped to the outside of each code safety valve and PORV tail pipe, as shown in Figure 1.4-14. Flow through the tail pipe, which constitutes positive indication that the valve is open, causes flow noise and pipe vibration (acoustical acceleration). The accelerometer produces a piezoelectric charge proportional to the acceleration; the charge is converted to a voltage by a remote charge converter mounted on the west wall of the pressurizer shield at the 107-foot elevation. The voltage is applied to the TEC valve flow monitor module located in the control room—above the east access door to the rear of the supervisory panels. (TEC is the manufacturer's acronym.) The valve flow monitor module processes the voltage signal and indicates the flow on a lighted bar graph display calibrated in 10 increments of full flow. Full flow is 1.0 on the indicator. The monitor module contains a signal processing channel and display for each monitored valve: PCV-455C, PCV-456, PCV-464, PCV-466, and PCV-468.

If any one of the five monitor channels detects a flow signal greater than 25% of the full flow signal, it triggers a common Pressurizer PORV and Safety Acoustic Monitoring alarm on control room panel SAF.

2.4 Pressurizer Relief Tank

If a PORV or code safety valve lifts, the steam, water, hydrogen and other gases in the PRZR flow to the pressurizer relief tank (PRT) through a common 12 inch discharge line that serves all five valves. The piping is arranged as shown on Figure 1.4-15. This line is connected to the 12 inch perforated sprarger that is installed just above the bottom of the tank's bottom.

Normally, the tank is partially filled with water at or near containment ambient temperature and contains a predominantly nitrogen atmosphere maintained at a pressure of 0.5 to 3 psig. The nitrogen is maintained at this pressure by a nitrogen pressure regulator and is designed to eliminate air in-leakage. Sparging nozzles located beneath the water surface, discharge steam into the water volume. The mixing that results, condenses and cools the discharged steam. A ¼-inch vent hole is drilled into the relief valve discharge line inside the PRT toward

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 34	Group #	1	
	K/A #	008K1.04	
OK	Importance Rating	3.3	

Knowledge of the physical connections and / or cause-effect relationship between the CCWS and the RCS, in order to determine sources(s) of RCS leakage into the CCWS

Proposed Question: Common 32

Unit 3 is at 100% power when the following events occur in the order shown:

- PRMS channels R-17A/17B, Component Cooling Water Activity Monitors, IN alarm.
- CCW Surge TANK levels increasing rapidly.
- Annunciator, RCP THERMAL BARRIER COOLING RETURN HIGH TEMP, in alarm.
- Pressurizer level decreases and the running charging pump speed goes to maximum.
- Annunciator, PRESSURIZER LOW LEVEL in alarm.

Which ONE of the following describes the event that has occurred?

A. A CVCS letdown non-regenerative tube has burst and LCV-459/460, High Press L/D Isol Valves, have failed to close.

B. A CVCS letdown non-regenerative tube has burst and protective functions have responded as designed.

C. A RCP thermal barrier leak has occurred and protective functions have responded as designed.

D. A RCP thermal barrier leak has occurred and MOV-625, RCP Thermal Barrier Outlet Valve, has failed to close.

Proposed Answer:

D. An RCP thermal barrier leak has occurred and MOV-625, RCP Thermal Barrier Outlet Valves, has failed to close.

Explanation (Optional):

Technical Reference(s): SD-4.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CCW001 0004, 0006 (As available)

Question Source: Bank # INPO 26954
Modified Bank # (Note changes or attach parent)
New

Question History: 12/15/2003 Turkey Point 3

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2,9
55.43

Comments:

813E for compressor 32 using local flow and temperature indicators FI-666 and 668 and TI-667 and 669. CCW is isolated to the waste gas compressors unless the compressor is in service. Flow is manually throttled at 25 to 30 gpm.

CCW also provides normal makeup to the seal tank and is automatically controlled by level control valves on the cooler inlets after CCW has been unisolated.

Relief valves 821C and D provide overpressure protection at 150 psig.

2.7.5 RCPs and Vessel Support Pads (Figure 4.1-6)

(Header 32)

One 6 inch supply line provides CCW flow to all four RCPs and the four Reactor Vessel Support Blocks. Normally, 20 gpm total is sent to the blocks and about 720 gpm to the RCPs. Each RCP requires approximately 5 gpm to the lower motor bearing cooler, 150 gpm to the upper motor bearing cooler, and 25 gpm to the thermal barrier. The throttle valves and flow indicators used to set these flowrates are all inside Containment.

There are two common return headers from the RCPs and the support blocks. One return header is for the upper and lower RCP bearing oil coolers and the support blocks. The other header is for the thermal barriers of all four RCPs.

The thermal barriers are segregated for leak considerations since a rupture at this point in the system would result in RCS inleakage to CCW. The portion of the CCW System from the inlet check valves to the containment isolation valves is rated for 2500 psig and 650°F.

Leak protection is provided by FCV-625 and FE-625. If return flow increases to 175 gpm, FCV-625 will close and isolate the thermal barriers. DP switch FIC-625 has a mechanical internal dampening feature set for 8 to 12 seconds. The dampening prevents spurious closure of FCV-625 due to the transients associated with sudden pressure surges when a CCW pump is secured or started, but allows closure of FCV-625 under sustained high flow conditions.

Overpressure protection is provided for each thermal barrier return line. These relief valves are set at 2485 psig.

In addition to leak protection, the thermal barrier return line flow transmitter provides Control Room indication and a low flow alarm at 100 gpm (Thermal Barrier CCW Header Low Flow on SGF). This alarm will actuate if FCV-625 closes. TIC-624, also on the thermal barrier return line, actuates a high temperature alarm at 140°F (RCP Thermal Barrier Cooling Return High Temp on SGF).

One temperature detector in the combined bearing cooler return actuates a high temperature alarm on Panel SGF at 130°F (RCP Bearings Cooling Water Return High Temp).

2.7.6 Excess Letdown Heat Exchanger (Figure 4.1-7)

(Header 32)

All valves for the excess letdown heat exchanger are located just outside Containment. Normally, no flow is supplied to the heat exchanger. When it is to be placed in service, only the Containment isolation valves have to be opened. These are controlled from the Control Room. When in service, approximately 240 gpm will be supplied.

2.7.7 RCP Seal Water Return Heat Exchanger (Figure 4.1-8)

(Header 32)

210 gpm is supplied to the RCP Seal Water Heat Exchanger. This is set using valve 808, a manual throttle valve. Outlet flow and temperature indicators, FI-605 and TI-606, are available locally.

2.7.8 CVCS Non-Regen Heat Exchanger (Figure 4.1-9)

(Header 32)

The flowrate through the Non-Regen Heat Exchanger (NRHX) is automatically controlled by TCV-130, a 6 inch ball valve. Letdown temperature is fed to temperature controller TIC-130 in the Control Room, which controls TCV-130 automatically. CCW outlet flow and temperature indication is available on FI-607 and TI-608.

Relief valve AC-812 relieves around TCV-130 if pressure exceeds 150 psig.

2.7.9 Gross Failed Fuel Detector (Figure 4.1-10)

(Header 32)

14 gpm is supplied to the detector. The flowrate is controlled manually with valve 1899B. Local flow indication is provided on FI-657.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 35	Group #	1	
	K/A #	010A2.02	
OK	Importance Rating	3.9	

Ability to predict the impacts of a spray valve failure on PZR PCS and on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations

Proposed Question: Common 33

Given the following:

- Unit 3 reactor power is 30%.
- RCS pressure is 2075 psig and slowly lowering.
- All Pressurizer heaters are energized.
- You notice that PCV-455B (PZR spray) is failed OPEN.
- When placed in manual PCV-455B will NOT close.
- Operators removed control power fuses for PCV-455B but valve will not close

Which ONE of the following is the proper sequence of actions to stop the pressure reduction?

A. Manually trip the reactor.
Trip 33 and 34 RCPs.
Go to E-0, Reactor Trip Or Safety Injection.

B. Trip 33 RCP.
The RCP trip will NOT cause a reactor trip at this power.
Dispatch an NPO to locally isolate Spray Valve PCV-455B.

C. Reduce Power to <25% so a RCP trip will NOT cause a reactor trip.
Trip 33 RCP.
Dispatch an NPO to locally isolate Spray Valve PCV-455B.

D. Trip 33 and 34 RCPs.

The reactor will trip when the RCPs are tripped.
Go to E-0, Reactor Trip Or Safety Injection.

Proposed Answer:

A. Manually trip the reactor.
Trip 33 and 34RCPs.
Go to E-0, Reactor Trip Or Safety Injection.

Explanation (Optional):

Technical Reference(s): 3-ARP-003, page 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # INPO 22876
Modified Bank # (Note changes or attach parent)
New

Question History: 12/9/2002 Cook 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

3.4.2 IF PRZR PORV(s) are NOT closed, THEN:

3.4.2.1 CLOSE PORV(s).

3.4.2.2 IF PORV(s) do NOT close, THEN:

a) CLOSE the affected PORV Block Valve.

b) REMOVE power fuses from the PORV solenoid to fail valve closed:

- RC-PCV-455C SOLENOID FUSES (Panel FCR, FU-398 & 399)
- RC-PCV-456 SOLENOID FUSES (Panel FBR, FU-347 & 348)

c) Dispatch NPO to DE-ENERGIZE the affected PORV Block Valve:

- RC-MOV-536 at MCC 36A
- RC-MOV-535 at MCC 36B

d) IF PRZR PORV(s) are NOT closed, THEN GO TO 3-AOP-LEAK-1, Sudden Increase in Reactor Coolant System Leakage.

3.4.3 IF PRZR Spray Valve(s) are NOT closed, THEN:

3.4.3.1 CLOSE PRZR Spray Valve in manual.

3.4.3.2 IF PRZR Spray Valve does NOT close, THEN CLOSE PRZR Spray Valve by removing power fuse from back of Foxboro controller for affected PRZR Spray Valve.

3.4.3.3 IF PRZR Spray Valve(s) can NOT be closed, THEN:

a) TRIP the Reactor.

b) TRIP 33 and 34 RCPs.

c) GO TO E-0, Reactor Trip or Safety Injection.

(CONTINUED ON THE NEXT PAGE)

The pressure indicator, PI-455 or PI-457, of the channel supplying the control train is illuminated on the control board.

2.7.3 PRZR Pressure Alarm Train

The alarm train consists of a dual high/low pressure comparator PC-456 F/G which compares the signal representing actual pressure to fixed high/low pressure setpoints, 2335 psig and 2185 psig, respectively. At 2335 psig, PC-456F triggers the pressurizer high-pressure alarm on control room panel SAF. This alarm function constitutes a backup for a component in the control train, PC-455I, which triggers the same alarm at 2310 psig.

At 2335 psig, PC-456F also sends a signal to the opening logic for RC-PCV-456. If PC-457F also senses a 2335 psig from interlocking channel PT-457, and the control switch for RC-PCV-456 is in AUTO, relief valve RC-PCV-456 opens.

Pressure bistable PC-456G triggers the Pressurizer Low Pressure alarm on panel SAF, at 2185 psig. This same alarm can be triggered by bistables PC-455J in the control train, also at 2185 psig.

2.7.4 PRZR Pressure Control Train

The PRZR pressure control train is composed of:

- pressure controller PC-455K,
- control heater group controller PC-455L,
- spray valve RC-PCV-455A controller, PC-455G,
- spray valve RC-PCV-455B controller, PC-455H,
- dual high/low pressure deviation comparator PC-455 I/J, and
- trip bistable for PORV RC-PCV-455C, PC-455F.

2.7.5 PRZR Pressure Controller

The control signal development is shown schematically in Figure 1.4-23. Actual pressure, designated P_{ACT}, is the input to the PRZR pressure controller PC-455K, located on panel FBF in the control room. A manually adjustable setpoint signal, designated P_{REF}, is set by means of a dial on the face of PC-455K. During normal operations, the setpoint is set to 2235 psig, which corresponds to a PC-455K setting of 66.8% (Refer to Attachment A for detailed discussion on PC-455K operation).

Controller PC-455K is a PID (Proportional, Integral, and Derivative) controller. The derivative function is set to zero. The proportional (P) component of PC-455K is functionally a summer with gain and bias. The summer receives P_{REF} as a negative input and P_{ACT} as a positive input; it determines the offset. It then imposes a gain of 2 on the offset, that is, the output signal is twice the size of the input signal (the offset). The proportional component also applies a bias that produces a 30 milliamp (ma) output signal when $P_{ACT} = P_{REF}$. Thus, the proportional component of PC-455K, P_{ERROR}, varies above or below 30 ma as P_{ACT} varies above or below P_{REF}.

The integral component of PK-455K adjusts the controller signal based on the length of time P_{ACT} differs from P_{REF}. In particular, the magnitude of the output signal increases as long as the offset is not zero.

This compensated P_{ERROR} signal is fed to:

- both spray valve controllers,
- the PRZR heater control group (modulating) controller and
- the deviation comparator (PC-455 I/J), which controls the backup heaters, and
- the trip bistable (PC-455F) for PORV RC-PCV-455C.

The operation of these components as part of the overall pressure control scheme is shown graphically in Figure 1.4-24. This scheme is based on:

- P_{REF} set to 2235 psig,
- all groups of backup heaters in AUTO, and
- both spray valves being set to operate at the same setpoints.

The assumption the P_{REF} = 2235 psig is useful because this is normal operating pressure. However, this can be changed on PC-455K, therefore the offset values are given in parentheses.

In addition, Figure 1.4-24, and the discussion below, do not include the effects of the integral (I) component of PC-455K. The actual output signal is greater than the arithmetic difference between P_{ACT} and P_{REF}, if that difference has existed (with the same sign) for a length of time. This means the system responses may occur at higher or a lower pressures than listed.

2.7.6 PRZR Heater Control Group Controller

The PRZR Heater Control Group (modulating) controller PC-455L, located in instrument rack (Foxboro) B-6 is a proportional controller with automatic and manual control modes. In automatic, the controller uses the compensated P_{ERROR} signal to generate the control signal for the control group heater SCRs. When pressure is at P_{REF} the output of PC-455L is at 50% of maximum and the thermal output of the control group is 50% of maximum. By the time pressure rises to 2250 psig (P_{REF} + 15 psi) the controller output is zero and the control group heaters are off. By the time pressure falls to 2220 psig (P_{REF} - 15 psi) the modulating heaters are fully on.

In manual, the compensated P_{ERROR} signal is disconnected from the controller and the control signal to the SCRs is adjusted using the manual operation bar on the controller.

2.7.7 PRZR Spray Valve Controllers

The spray valve controllers PC-455G and PC-455H, located on panel FBF, are proportional (P) controllers with automatic and manual control modes. In automatic, they use the compensated P_{ERROR} to generate the control signal for positioning the spray valves. They generate no output until pressure rises to 2260 psig (P_{REF} + 25 psi). Then they ramp open the spray valves in a linear fashion until they are full open at 2310 psig (P_{REF} + 75 psi).

In manual, the compensated P_{ERROR} signal is disconnected from the spray controller. The signal to the spray valves is adjusted using the manual operation bar on the controller.

The compensated P_{ERROR} signal is applied to a high-pressure comparator, PC-455F, which at 2335 psig (P_{REF} + 100 psi) supplies a trip signal to the opening logic for PORV, RC-PCV-455C. The PORV opens if the interlock signal from PC-474B is also present and its control switch is in the auto position. The open permissive input to RC-PCV-455C ensures that the integrating (I) function of PC-455K does not cause RC-PCV-455C to open at a pressure less than 2335 psig. However, the integrating function may delay opening of PCV-455C until pressure is greater than 2335 psig.

Finally, the compensated P_{ERROR} signal is applied to the dual high/low pressure comparator PC-455 I/J. AT 2310 psig (P_{REF} + 75 psi), PC-455I triggers the Pressurizer High Pressure annunciator on panel SAF. (PC-456F in the alarm train triggers this same alarm window at 2335 psig fixed setpoint.) At 2185 psig (P_{REF} - 50 psi), PC-455J triggers the Pressurizer Low Pressure annunciator on panel SAF. This is the same 2185 psig alarm (fixed setpoint) that PC-456G annunciates. Also, at 2185 psig (P_{REF} - 50 psi) PC-455J energizes the backup heaters that are selected to AUTO. The backup heaters remain on until pressure is restored to 2200 psig (P_{REF} - 35 psi), which resets PC-455J and turning the backup heaters off.

The following is a table of error signal/control functions based on the output of the master controller (PC-455K) (without the integral component).

PRZR Functions as a Function of PC-455K Output

Function	Pressure Deviation from setpoint (psi)	PC-455K Output (%)
Deviation Low Press Alarm	-50	37.50
Backup Heaters On	-50	37.50
Backup Heaters Off	-35	41.25
Control Heaters Full On	-15	46.25
Pressure Setpoint (P_{REF})	0	50.00
Control Heaters full Off	+15	53.75
Spray Initiation	+25	56.25
Spray Full On	+75	68.75
High Pressure Alarm	+75	68.75
PORV Closes	+85	72.50
PORV Opens	+100	75.00

2.7.8 PRZR Pressure Control System Operation

The current mode of operation of the PRZR pressure control system differs from that described above.

The first difference is that one spray valve controller, either PC-455G or PC-455H, is biased such that its associated valve opens before (or leads) the other. This is done to produce a smoother spray initiation.

The second difference is that one set of backup heaters is kept on. Continuously energizing one set of backup heaters increases the heat input to liquid space, raising the pressure. The pressure increase causes:

- Control heater group thermal output to decrease (Off at 2250 psig (P_{REF} + 15 psi)), and
- Spray valves to open (2260 psig (P_{REF} + 25 psi)).

The pressure increase caused by the backup heaters stops when it is balanced by the condensation caused by the additional spray flow. However, the system cannot remain at this pressure (above 2260 psig) because this produces a non-zero offset in PC-455C. As long as a non-zero offset exists, the integral (reset) component of PC-455C causes the magnitude of the control signal to continue to increase. Thus, the spray valves receive a signal to open more. The additional spray flow causes pressure to decrease toward 2235 psig. However, if the offset-plus-reset signal decreases to the point that spray valves close, the energized backup heaters raise pressure above 2235 psig again, the offset-plus-reset signal increases and a spray valve eventually reopens.

The system can only achieve stability (a relatively constant pressure) when:

- the pressure increase caused by the heat input of the backup heaters

equals

- the pressure reduction caused by the total spray flow (bypass flow plus spray valve(s) flow).

Since the bypass spray flow is insufficient to compensate for the energized backup heater group, the spray valve(s) remain open (or continually cycle open) a small amount. This requires a relatively continuous open demand signal from the PRZR pressure controller. The demand signal is due to the integration (reset) signal developed from of a small offset above 2235 psig. Therefore, with one set of backup heaters ON, the RCS pressure stabilizes with a spray valve open a small amount, at a pressure above 2235 psig, but less than 2260 psig.

2.8 Over-pressurization Protection System

The Over-pressurization Protection System (OPS) is designed to prevent over-pressurization of the reactor vessel when the RCS is at low temperatures. At low temperatures, the reactor vessel is less ductile and more susceptible to brittle fracture. Therefore, stress on the vessel from all sources must be reduced as the vessel temperature decreases. In particular, the maximum allowable RCS pressure must be reduced, as a function of temperature, to limit the pressure-induced

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 36	Group #	1	
	K/A #	012A1.01	
OK	Importance Rating	2.9	

Ability to predict and / or monitor changes in parameters to prevent exceeding design limits associated with operating the RPS controls including trip setpoint adjustments

Proposed Question: Common 34

During the performance of an NIS power range heat balance at 90% power, an operator uses a feedwater temperature 30°F lower than actual. Would the calculated value of power be HIGHER or LOWER than actual power, and would an adjustment of the NIS power range channels, based on this value, be CONSERVATIVE or NON CONSERVATIVE with respect to High Power Reactor Trip protection setpoints?

Calculated Power Setpoints would be…

A. Higher - Non Conservative

B. Higher - Conservative

C. Lower - Non Conservative

D. .Lower - Conservative

Proposed Answer:

B. Higher - Conservative

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # INPO 27669
Modified Bank # ____________ (Note changes or attach parent)
New ____________

Question History: 10/5/2004 Cook 1

Question Cognitive Level: Memory or Fundamental Knowledge ____________
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ____________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 37	Group #	1	
	K/A #	013A1.09	
OK	Importance Rating	3.4	

Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ESFAS controls including T-hot.

Proposed Question: Common 35

Initial Conditions:

- The plant is at 100% power, beginning of life.
- Rod Control is in MANUAL.
- Tave is on program.
- The Reactor Engineer has requested the crew to slowly withdraw control bank "D" rods to full out after Moderator Temperature Coefficient (MTC) testing.
- The crew is to allow MTC to control reactor power, without borating during the rod withdrawal.

The RO slowly withdraws control bank "D" rods, resulting in the following:

- RCS Narrow Range Thot increases by 4°F.
- PZR pressure control system maintains RCS pressure stable.
- Delta Flux remains in the program band.

How do the OTΔT and OPΔT trip setpoints respond?

A. OTΔT setpoint DECREASES.
OPΔT setpoint DECREASES.

B. OTΔT setpoint DECREASES.
OPΔT setpoint DOES NOT CHANGE.

C. OTΔT setpoint DOES NOT CHANGE.
OPΔT setpoint DECREASES.

D. OTΔT setpoint DOES NOT CHANGE.
OPΔT setpoint DOES NOT CHANGE.

Proposed Answer:
A. OTΔT setpoint DECREASES.
OPΔT setpoint DECREASES.

Explanation (Optional):

Technical Reference(s): ITS 3.3.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICRXR E1 (As available)

Question Source: Bank # INPO 27099
Modified Bank # (Note changes or attach parent)
New

Question History: 7/16/2004 Millstone 3

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

Table 3.3.1-1 (page 7 of 8)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.8% of ΔT span :

$$\Delta T \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, ≤ [*]°F.

P is the measured pressurizer pressure, psig
P′ is the nominal RCS operating pressure, ≤ [*] psig

$K_1 \leq$ [*]	$K_2 \geq$ [*]/°F	$K_3 \geq$ [*]/psig
$\tau_1 \geq$ [*] sec	$\tau_2 \leq$ [*] sec	

$f_1(\Delta I)$ =	[*] {[*] + $(q_t - q_b)$}	when $q_t - q_b \leq$ - [*]% RTP
	0% of RTP	when -[*]% RTP < $q_t - q_b \leq$ [*]% RTP
	-[*] {$(q_t - q_b)$ - [*]}	when $q_t - q_b$ > [*]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with [*] are specified in the COLR.

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

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.8% of ΔT span:

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{(1+\tau_3 s)} T - K_6 \left(T - T''\right) - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T" is the nominal T_{avg} at RTP, ≤ [*]°F.

$K_4 \leq$ [*]

$K_5 \geq$ [*] /°F for increasing T_{avg}
[*] /°F for decreasing T_{avg}

$K_6 \geq$ [*] /°F when T > T"
[*] /°F when T ≤ T"

$\tau_3 \leq$ [*] sec

$f_2(\Delta I)$ = [*]

*The values denoted with [*] are specified in the COLR.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 39	Group #	1	
	K/A #	022K2.01	
OK	Importance Rating	3.0	

Knowledge of power supplies to the containment cooling fans

Proposed Question: Common 36

A small break LOCA occurred causing a Reactor Trip AND Safety Injection. All safeguards equipment operated as designed. During the performance of E-0, Reactor Trip or Safety Injection, Offsite Power was lost.

Given the following:

- 31 EDG failed to start
- All remaining equipment operated as designed

All equipment that should have automatically started has started on their respective buses. With no operator action, what will be the configuration for the Containment Fan Cooler Units (FCU)?

A. 31, 32, 33 and 35 FCU's running

B. 31, 33 and 35 FCU's running

C. 33, 34 and 35 FCU's running

D. 32, 33, 34 and 35 FCU's running

Proposed Answer:
B. 31, 33 and 35 FCU's running

Explanation (Optional):

Technical Reference(s): SD-10.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-VCCARC 0003 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

EMERGENCY DIESELS ONE-LINE DIAGRAM

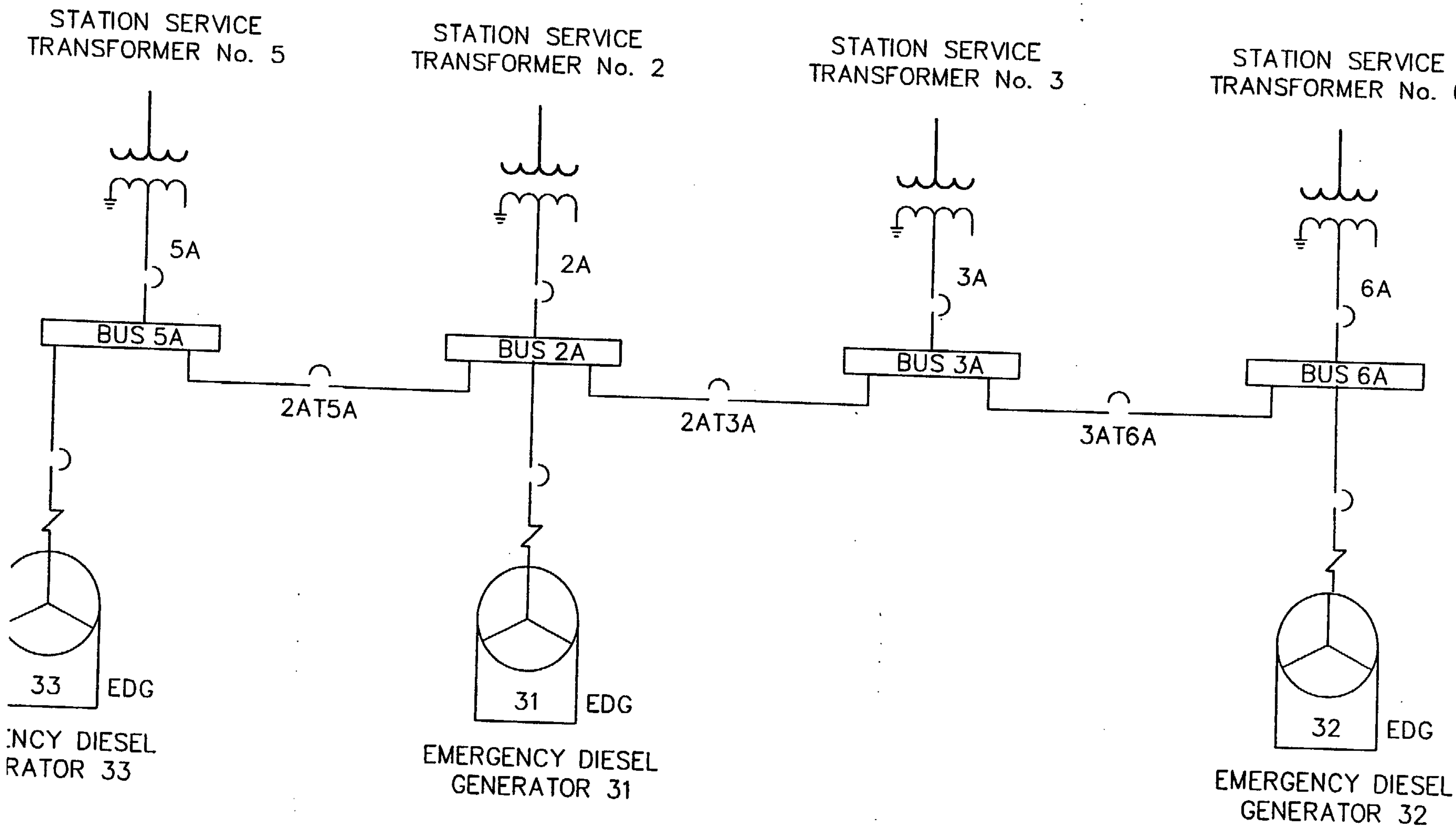


Figure 27.3-1: Emergency Diesels One-Line Diagram (EDS_01)

WS 39

Related Power Supplies

Component	Power Supply	Comments
31 Fan Cooler Unit	480 V Bus 5A	
32 Fan Cooler Unit	480 V Bus 2A	
33 Fan Cooler Unit	480 V Bus 5A	
34 Fan Cooler Unit	480 V Bus 3A	
35 Fan Cooler Unit	480 V Bus 6A	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 40	Group #	1	
	K/A #	026K3.01	
OK	Importance Rating	3.9	

Knowledge of the effect that a loss or malfunction of the CSS will have on the CCS

Proposed Question: Common 37

Given the following conditions:

- Unit 3 is operating at 100% power.
- 31 Containment Spray pump has been declared INOPERABLE due to an oil leak.
- 32 and 34 Fan cooler Units (FCU) are INOPERABLE and isolated due to service water leaks.
- All other ECCS equipment is OPERABLE.

With the plant in this configuration, which of the following describes if the plant is being operated within the Design Basis for containment cooling, and the BASES for your answer?

A. No, two (2) Containment Spray pumps and five (5) FCU's are required to be OPERABLE to meet the design basis for containment cooling.

B. No, one containment Spray pump and four (4) FCU's are required to be OPERABLE to meet the design basis for containment cooling.

C. Yes, one (1) OPERABLE Containment Spray pump combined with three (3) OPERABLE FCU's meets the design basis for containment cooling.

D. Yes, a single OPERABLE Containment Spray pump meets the design basis for containment cooling.

Proposed Answer:

C. Yes, one (1) OPERABLE Containment Spray pump combined with three (3)

OPERABLE FCU's meets the design basis for containment cooling.

Explanation (Optional):

Technical Reference(s): SD-10.3
ITS 3.6.6
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-VCCARC 0004, 0006 (As available)

Question Source: Bank # INPO 24076
Modified Bank # (Note changes or attach parent)
New

Question History: 5/5/2003 Salem Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Specification 4.5.A.4 specifies testing requirements for the containment air filtration system.

3.5 Improved Technical Specifications

TS 3.6.5 (Containment Air Temperature) limits containment average air temperature during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss of Coolant Accident (LOCA) or Steam Line Break (SLB). The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient and has been determined to be acceptable for the DBA LOCA or SLB. The bases of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment.

TS 3.6.6 (Containment Spray System and Containment Fan Cooler System) sets limits on FCU and containment spray trains. Accident Analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a. Two containment spray trains; or,

b. Three fan cooler trains (i.e.., five fan cooler units); or

c. One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

The last configuration, one containment spray train and two fan cooler trains, is the minimum configuration available following the loss of any safeguards power train (e.g., diesel failure).

TS 3.4.15 (RCS Leakage Detection Instrumentation) requires instruments of diverse monitoring principles operable to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS leakage indicates possible Reactor Coolant Pressure Boundary (RCPB) degradation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 29	Group #	1	
	K/A #	003K6.04	
OK	Importance Rating	2.8	

Knowledge of the effects of a loss or malfunction on the containment isolation valves affecting RCP operation will have on the RCPs

Proposed Question: Common 38

Unit 3 was operating at 100% power when an inadvertent Phase "B" isolation occurred. Which of the following describes the required actions for the Phase "B" isolation and the reason for performing those actions?

A. Immediately trip the Reactor, trip all four RCPs and enter E-0, Reactor Trip due to loss of RCP seal return flow.

B. Immediately trip the Reactor, trip all four RCPs and enter E-0, Reactor Trip due to loss of CCW to all four RCPs.

C. If Phase "B" CCW supply and return valves for RCP motor cooling are not opened within 2 minutes then trip the Reactor, trip all four RCPs and enter E-0, due to loss of RCP motor cooling.

D. If Phase "B" CCW supply and return valves for RCP motor cooling are not opened within 2 minutes then trip the Reactor, trip all four RCPs and enter E-0, due to loss of RCP seal cooling.

Proposed Answer:

C. If Phase "B" CCW supply and return valves for RCP motor cooling are not opened within 2 minutes then trip the Reactor, trip all four RCPs and enter E-0, due to loss of RCP motor cooling.

Explanation (Optional):

Technical Reference(s): 3-AOP-CCW-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

LOS #29

4. SUBSEQUENT ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.1 __Are <u>any</u> CCW pumps operating?	__ Start an available CCW pump.
4.2 Do <u>any</u> of the following conditions exist? __System leak __Low system flow/pressure __High system temperature	__ **RETURN** to procedure and step in effect.
4.3 __**IAAT** either of the following conditions exist: __CCW flow to <u>any</u> RCP is lost for ≥ 2 minutes __RCP motor bearing temperature exceeds 200°F **THEN** perform Steps 4.4 - 4.7.	__ **GO TO** Step 4.8.
4.4 __Are the Reactor Trip Breakers closed?	1. __Trip affected RCPs. 2. __**GO TO** Step 4.8.
4.5 __Trip the reactor.	
4.6 __Trip affected RCPs.	
4.7 __**INITIATE** E-0.	
4.8 **GO TO** applicable step based on condition indicated (in order of priority):	

√	CONDITION	STEP
	System leak	4.9
	Low system flow/pressure	4.74
	High system temperature	4.108

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 42	Group #	1	
	K/A #	039K5.08	
OK	Importance Rating	3.6	

Knowledge of the operational implications of the effect of steam removal on reactivity as it applies to the MRSS

Proposed Question: Common 39

Given the following conditions:

- A Unit startup is in progress following a mid-cycle outage.
- The reactor is critical at 1E-8 amps.
- A condenser steam dump valve fails partially open.

Assuming NO action by the operating crew, which one of the following describes the immediate effect on the plant?

A. RCS Temperature INCREASES; Power INCREASES.

B. RCS Temperature INCREASES; Power DECREASES.

C. RCS Temperature DECREASES; Power DECREASES

D. RCS Temperature DECREASES; Power INCREASES.

Proposed Answer:
D. RCS Temperature DECREASES; Power INCREASES.

Explanation (Optional):

Technical Reference(s): ______ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: TAA-C-005 2504 (As available)
TAA-C-011 02532

Question Source: Bank # INPO 24963
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 44	Group #	1	
	K/A #	059K4.08	
OK	Importance Rating	2.5	

Knowledge of MFW design features(s) and / or interlock(s) which provide for feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch)

Proposed Question: Common 40

Given the following conditions:

- A plant startup is in progress
- The Unit is at 30% power.
- All Main Feed Regulating Valves are in AUTO
- 31 Feed Flow channel FT-418B is selected for control of 31 SG.
- 31 Feed Flow transmitter PT-418B fails 10% high.

Assuming NO operator action, which of the following statements describes the response of 31 Main Feed Reg (MFR) Valve?

A. 31 MFR valve will initially throttle in the CLOSE direction and then over time will return to it's original position.

B. 31 MFR valve will initially throttle in the OPEN direction and then over time will return to it's original position.

C. 31 MFR valve will CLOSE and then over time the Reactor will trip on Low SG level.

D. 31 MFR valve will OPEN and then over time the Turbine will trip on High SG level.

Proposed Answer:

A. 31 MFR valve will initially throttle in the CLOSE direction and then over time will return to it's original position.

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICLOVE E
I3LP-ILO-ICSGL 2.0 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS 44

STEAM GEN # 31
HIGH LEVEL

INPUT DEVICE: LC-417A
LC-417C
LC-417E

SETPOINT: 75% of span

1.0 CAUSES

1.1 High level on 1 of 3 SG level channels.

2.0 AUTOMATIC ACTIONS

2.1 IF level is equal to or greater than 75% on 2 of 3 level channels, THEN:

- Turbine trips
- MBFP discharge valves close

3.0 SUBSEQUENT ACTIONS

3.1 VERIFY alarm by observing all SG level indicators and trip status lights.

3.2 IF channel failure has occurred, THEN GO TO ONOP-RPC-1 [3AOP-INST-1], Instrument Failures.

3.3 IF level is high on only 1 of 3 level channels, THEN:

3.3.1 TRANSFER SG level control to manual as required.

3.3.2 RETURN level to programmed value.

(CONTINUED ON THE NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 45	Group #	1	
	K/A #	061K5.01	
OK	Importance Rating	3.6	

Knowledge of the operational implications of the relationship between AFW flow and RCS heat transfer as it applies to the AFW

Proposed Question: Common 41

Given the following plant conditions:

- A reactor trip has occurred from 100% power
- The operators have not operated any controls post-trip.
- The crew has just entered ES-0.1 Reactor Trip Response.
- PZR level is 25% and slowly decreasing.
- Steam Generator pressures are approximately 990 psig and slowly decreasing.
- Tave is 545°F and slowly decreasing.
- RCS pressure is 2020 psia and slowly decreasing.

What action must be taken by the crew per ES-0.1 to address the cooldown?

A. Commence immediate boration.

B. Throttle Auxiliary Feedwater flow.

C. Initiate SI and return to step 1 of E-0.

D. Close the MSIVs and MSIV bypass valves.

Proposed Answer:

B. Throttle Auxiliary Feedwater flow.

Explanation (Optional):

Technical Reference(s): ES-0.1, Step 1 RNO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # INPO 27090

Modified Bank # (Note changes or attach parent)

New

Question History: 7/16/2004 Millstone 3

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 2,9

55.43

Comments:

WS 45

Number:	Title:	Revision Number:
ES-0.1	REACTOR TRIP RESPONSE	18

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. * CHECK RCS Average Temperature: *

   - ANY RCP running - RCS AVERAGE TEMPERATURE STABLE AT OR TRENDING TO 547°F

     OR

   - NO RCP running - RCS COLD LEG TEMPERATURES STABLE AT OR TRENDING TO 547°F

PERFORM the following:

a. IF temperature is less than 547°F AND decreasing, THEN PERFORM the following:

   1) STOP dumping steam.

   2) IF cooldown continues, THEN PERFORM the following:

      a) CONTROL total feed flow.

      b) MAINTAIN greater than 365 gpm flow until at least one SG NR level greater than 9%.

   3) IF cooldown continues, THEN PERFORM the following:

      a) CLOSE all MSIVs.

      b) IF MSIV(s) can NOT be closed, THEN DISPATCH NPO to locally close MSIVs per SOP-ESP-1.

      c) DISPATCH NPO to ensure all MSIV bypass valves are closed, if required.

(STEP 1 CONTINUED ON NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 38	Group #	1	
	K/A #	013K3.03	
OK	Importance Rating	4.3	

Knowledge of the effects that a loss or malfunction of the ESFAS will have on the containment

Proposed Question: Common 42

Given the following conditions:

- A Large Break LOCA has occurred.
- Train B ECCS has failed to actuate.
- All other actuations actuate and Train A ECCS equipment is running as required.

Assuming no action by the crew, which ONE (1) of the following describes the effect on the plant?

A. Containment Isolation Phase A will actuate. Phase B will NOT actuate.

B. Containment Isolation Phase A will NOT actuate. Phase B will actuate.

C. Containment Isolation Phase A will NOT actuate. Phase B will NOT actuate.

D. Containment Isolation Phase A and B will actuate.

Proposed Answer:

D. Containment Isolation Phase A and B will actuate.

Explanation (Optional):

Technical Reference(s): ______ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # INPO 28065
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: 9/27/2004 Robinson 2

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

The Containment pressure can be maintained less than the designed pressure of 47 psig and 271°F following the LOCA by:

- All five Containment Recirculation Fan Cooler Units,

Or,

- Both Containment Spray Pumps,

Or

- Three Containment Recirculation Fan Cooler Units and one Containment Spray Pump.

The spray additive tank contains a minimum of 4,000 gallons of solution with a sodium hydroxide concentration not less than 35% and no greater than 38% by weight. This volume, when mixed with the water from the RWST, accumulators and the Reactor Coolant System, will result in a solution in the Containment sump with a pH greater than 8.3. This allows continued iodine removal by spray during the recirculation phase of operation.

The RWST contains a minimum of 342,200 (35.4') gallons of water with a boron concentration >2400 but <2600 PPM. Approximately 167,000 gallons is used by the Containment Spray System, on a design basis accident.

In the first twenty minutes or so following the maximum LOCA, the Containment Spray Pumps operate to meet the design heat removal capacity for Containment. The total heat absorption capability of each spray pump delivering 2500 gpm is 2.18×10^8 BTU/HR based on addition of 100°F RWST water.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 41	Group #	1	
	K/A #	026A3.01	
OK	Importance Rating	4.3	

Ability to monitor automatic operation of the CSS including pump starts and correct MOV positioning

Proposed Question: Common 43

Given the following plant conditions:

- A Large Break LOCA occurred with SI actuation
- 15 seconds after the SI actuation containment pressure rises to 24 psig

Which one of the following sets of pumps/valves receives a start/open signal on the automatic Containment Spray Actuation?

A. Spray pumps 31 and 32 after ~34 second time delay, MOV 866A and B, CNMT Spray Pump Discharge Valves immediately and AOV 876A and B, CNMT Spray NaOH addition after ~2 minute time delay.

B. Spray pumps 31 and 32 after ~34 second time delay, 869A and B, Containment Spray Header Isolation Valves immediately, and MOV 866A and B, CNMT Spray Pump Discharge Valves immediately.

C. Spray pumps 31 and 32 after ~2 minute time delay, 869A and B, Containment Spray Header Isolation Valves, and 880A-K, CNMT Spray Charcoal Filter Douse Valves.

D. Spray pumps 31 and 32 after ~2 minute time delay, AOV 876A and B, CNMT Spray NaOH addition and 880A-K, CNMT Spray Charcoal Filter Douse Valves.

Proposed Answer:

A. Spray pumps 31 and 32 after ~34 second time delay, MOV 866A and B, CNMT Spray Pump Discharge Valves immediately and AOV 876A and B, CNMT Spray

NaOH addition after ~2 minute time delay.

Explanation (Optional):

Technical Reference(s): SD-10.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CS001 E-1 (As available)

Question Source: Bank # INPO 27300
Modified Bank # (Note changes or attach parent)
New

Question History: 4/27/2004 Ginna 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

contents of the RWST. At 1.5 ft in the RWST the operator secures the spray pump and shuts the discharge valve 866A.

The Containment Spray Headers are supplied by the recirculation pumps during the recirculation phase. This water is from the recirculation sump, cooled to approx. 134°F through the residual heat exchanger, and results in a heat removal rate of 1.63 x 10^8 BTU/HR. This equals the core decay heat after 5000 seconds. The recirculation phase of Containment Spray continues following the LOCA to ensure the complete removal of iodine from the Containment atmosphere. The recirculation mode of operation is detailed in System Description 10.1 Safety Injection.

The spray system is automatically actuated on a high high Containment pressure of approx. 22 psig. This signal is a coincident signal (2/3) from two separate channels. Each channel consists of three pressure instruments. One transmitter from each channel comes from a common containment sensing line. Three sensing lines feed six transmitters. One channel is feed from PI-948A, B, C and the other channel from PI-949A, B, C. The actuation of either channel, energized to trip, causes an alarm on the safeguards panel (SBF-2) and indicates on the trip status panel (SOF) which channel has tripped. Both channels (2/2) must trip to actuate the system. Redundant logics are used in the formation of the high-high Containment pressure signal to prevent actuation of the Containment Spray System on a spurious signal or loss of power. This signal acts as a back up to the Safety Injection signal and results in a phase "B" Containment isolation, a Containment Ventilation isolation, and a Steam Line isolation.

The system can be manually actuated by simultaneously depressing two red Spray Initiation pushbuttons on safeguards panel SBF-1 in the control room. This manual action also initiates a Containment Ventilation isolation and a Containment phase "B" isolation. The manual signal does not activate a Safety Injection signal or the Steam Line isolation signal. The Containment Spray actuation logics are detailed in System Description 28, Overall Unit Protection.

When the Containment Spray System is actuated, each motor operated pump discharge valve (866A and 866B) opens and the spray pumps start. A portion of each pump discharge is directed through two liquid jet eductors which add the sodium hydroxide solution contained in the spray additive tank into the spray pump suction.

The spray additive tank remains isolated for two minutes after a Containment Spray actuation signal. This two minute time delay gives the operators time to assess plant conditions and to cancel the Sodium

Hydroxide addition. Isolation valves (876A and 876B) for spray additive tank are normally shut, fail open, air operated diaphragm valves. The isolation valves can be opened by switches located on SBF-1 panel in control room. Automatic opening of the spray additive tank isolation valves on Containment Spray signal can be canceled by pushing the red NaOH cancel pushbutton on SBF-1 panel in control room. (Currently, there is no procedural guidance to cancel NaOH addition even if the signal is inadvertent.)

The Containment Spray actuation signal is reset by the use of two pushbuttons, one for each logic channel, located on the safeguards panel SBF-1. Depressing both pushbuttons will not stop any running equipment, but it will permit the operator to take manual action to secure the spray pumps and shut the motor operated discharge valves. If Containment pressure again reaches the actuation pressure of approx. 22 psig, the Containment Spray System will again actuate.

The system also supplies fire-dousing water to the five Containment Recirculation Fan Cooler Unit charcoal filter banks. The water-dousing supply is used in the unlikely event of a fire resulting from the heat generated by the decay of iodine absorbed in the charcoal filters. The filter banks of each fan unit are supplied through two, normally shut motor operated valves. These valves are operated with individual, two-position switches, and individual valve position is indicated with red (open) and green (shut) lights all located on panel SMF in the control room.

2.2 Containment Spray Pumps 31 and 32

The Containment Spray Pumps are located at elevation 41 foot in the Primary Auxiliary Building. The pumps are single stage horizontal centrifugal pumps rated for 2600 gpm at a 427 foot discharge head. The pumps are equipped with mechanical seals to eliminate leakage and require no external means of cooling. They are driven by 400 HP electric motors powered from 480V buses 5A and 6A, respectively.

The spray pumps are designed to deliver their rated flow with the RWST at a level of 1.5 feet (empty) against a head equal to the sum of the design pressure of Containment, the head to the uppermost spray nozzles, and the line and nozzle pressure losses. Each spray pump provides 50% of the design flow to the Containment Spray Headers necessary to maintain Containment pressure below 47 psig.

The pumps are controlled using individual, four-position switches located on the SBF-1 panel. The switch positions are PULL-OUT,

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 46	Group #	1	
	K/A #	061K6.02	
OK	Importance Rating	2.6	

Knowledge of the effect of a loss or malfunction of the pumps will have on the AFW components

Proposed Question: Common 44

Given the following plant conditions:

- Natural Circulation C/D in progress at 20°F/hr
- S/G Atmospheric Steam dumps in manual for C/D
- 32 ABFP supplying 125 gpm to each of the four S/Gs
- 31 and 33 ABFPs are shutdown in AUTO
- All S/G levels being maintained at 45%

What would be the effect on the AFW System should 32 ABFP trip on overspeed? (Assume NO operator actions)

A. No AFW Pump would start causing all four SGs to eventually dry out.

B. Both motor driven AFW Pumps would immediately start and commence feeding all four SGs causing S/G levels to continually increase.

C. Both motor driven AFW Pumps would start when any one of the four S/G levels decreased to 8% causing SG levels to continually increase.

D. Both motor driven AFW Pumps would start when any one of the four S/G levels decreased to 8% causing SG levels to go to and automatically maintain program value.

Proposed Answer:

C. Both motor driven AFW Pumps would start when any one of the four S/G levels decreased to 8% causing SG levels to continually increase.

Explanation (Optional):

Technical Reference(s): SD-21.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-AFW001 0005 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

The ABFP motors are limited to 400 HP. The nominal voltage of their emergency power system limits the motor size to 400 HP. If flow increases to greater than 415 gpm the pump will trip on overcurrent. (Refer to DBD-303, section 3.1.3.7)

2.4.1 Auto Starts

The motor-driven ABFPs start automatically on the following signals (Refer to Figure 21.2-9):

- Loss of voltage on the 480V buses 6A (ABFP 33) or bus 3A (ABFP 31). The ABFPs will start 28 seconds after the undervoltage condition is clear. (Time delay relays in the starting circuits provide this time delay.)
- Low-Low level (8%) on 2/3 detectors on any steam generator.
- Automatic trip of either Main Boiler Feedwater Pump
- A Safety Injection Signal
- AMSAC Actuation Signal generated.

If the pump is turned off with an auto start signal present, it will restart in 28 seconds unless in the pull out position.

2.4.2 Motor-driven ABFP Trips

ABFP #31 and #33 breakers will trip on the following conditions:

- Overcurrent
- Low Suction flow when recirc valves control switch is in auto.
  - ♦ If flow is less than 40 gpm a flow switch will cause the recirculation valve for the effected pump to open. If flow is not greater than 75 gpm within 15 seconds after recirc valve is given open signal, the pump will trip. This trip is disabled when the recirculation valve control switch is in "Open" or "Close".
- Bus Strip conditions (Refer to Figure 21.2-8)
  - ♦ ABFPs #31 & #33 would be deenergized on undervoltage to their respective buses.

2.4.3 Motor Driven ABFP Controls

Each motor-driven ABFP has a control switch on panel SCF in the Control Room. The switch positions are:

- START – AFW pump will start if power is available

- Turning the supply (upper) valve 180 degrees bypasses the positioner and directs the actuation air to the lower valve.
- Operating the bypass (lower) valve will port control pressure to the valve diaphragm.

FCV-406A has a replacement positioner. It lacks the capability to manually apply I/A pressure to the valve diaphragm. Although it has the "rabbit ears" on the positioner, they are not fully functional.

All the Auxiliary Feed Regulating Valves may also be operated locally by failing the air and using the attached handwheel.

All eight valves have nitrogen backup to instrument air.

Each Auxiliary Feed Regulating Valve has a locked open inlet and outlet isolation valve and a non return check valve between the outlet isolation valve and the Auxiliary Feed Regulating Valve.

2.6.1 Motor-driven pump runout protection

The Auxiliary Feed Flow Regulating Valves in the motor-driven pump headers, FCV-406A, B, C, D are provided with a feature that prevents a motor-driven pump runout condition.

Runout can occur when the pumps are supplying feed to steam generators at any pressure. Operating at SG pressure reduced to as low as 110 psig presents the most limiting pump conditions. Runout could result in overheating and damage to the pump motors. High flow rates require high power output from the ABFW pump motors. Power greater than 400 hp reduces the service factor of the pump, decreasing its reliability. Limitations of the 480 VAC power supplies and their backup diesel generators require strict pump motor current limitations on the motor driven pump.

Pressure transmitter PT-406A(B) signal, at the discharge of each individual pump , is fed to its corresponding Pressure (cutback) Controller PC-406A(B). Setpoint of this controller is set biased on the total flow output signals from two SG ABFW supply line flow transmitters FT-1200 & 1201 (FT-1202 & 1203) Currently , the Pressure Controller setpoints are adjusted to obtain an ABFW flow rate of approximately 370 to 380 gpm per each pump.

This flow range was selected to provide a margin above minimum required AFWS flow of 345 gpm at the lowest bus voltage conditions. If the Pressure Controller input signal is <setpoint, the controller output to the HI Signal Selectors increases.

The Hi Selectors PM-406A(B) select the highest signal of either Pressure (cutback) Controller or the operators manual Hand

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 47	Group #	1	
	K/A #	062K2.01	
OK	Importance Rating	3.3	

Knowledge of bus power supplies to major system loads

Proposed Question: Common 45

The Main Generator just tripped due to a pilot wire transfer trip from Buchanan. The 25X1 sync check relay which ensures synchronization between 6.9 KV buses 5 and 1 for the auto transfer has failed. All other circuits are intact. Which of the following describes the affect to the Reactor Coolant Pumps (RCP)?

A. Only 32, 33, and 34 RCPs will be operating

B. Only 32 and 33 RCPs will be operating

C. Only 33 and 34 RCPs will be operating

D. Only 31, 32 and 33 RCPs will be operating

Proposed Answer:

B. Only 32 and 33 RCPs will be operating

Explanation (Optional):

6.9 KV bus tie 2-5 utilizes sync check relay between bus 1 and 5 also (25X1). Bus two will not transfer of offsite power. Bus 1 – 31 RCP. Bus 2 – 34 RCP.

Technical Reference(s): __________ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSRCP D (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

The pumps are operated from CCR panel SAF. RCPs #31, #32, #33 & #34 are powered from 6.9 kV buses 1, 4, 3 and 2 respectively.

RCPs are tripped on underfrequency and undervoltage conditions. Their setpoints and logic are found in System Description #28, Overall Unit Protection.

2.3.8 Vibration and Noise Indication

Two vibration detectors located in the lower motor section are routed to a remote readout panel in the lower electrical tunnel where a measuring device is plugged in to provide measurement. Maximum displacement is 0.002 inch (2 mils), peak to peak.

A complete system is also employed to continuously monitor motor vibrations of both shaft and frame. Two proximity probes per pump are mounted 90 degrees apart near the shaft. One mounted vertically in line with the pump discharge and the other mounted horizontally which is perpendicular to the discharge. Two frame mounted velocity seismoprobes mounted 90 degrees apart are also installed for frame monitoring. The proximitors condition the probes electrical input as well adjusting the return signal to provide an output proportional to probe-shaft gap. The velocity to displacement converters linearize the output from the seismoprobes.

A third probe (Keyphazer) is configured midway between the proximity probes. Sensing a notch in the coupling, it supplies a reference signal used to balance the pump.

Two meters per pump are located next to CCR rack C10. One meter reads shaft vibration (0 to 30 mils) while the other reads frame vibration displacement (0 to 10 mils). The meters read the highest probe and each has a switch enabling the selection of either probe.

The monitors contain alert and danger alarm relays adjustable over the full range of scale and annunciate a common alarm on CCR panel SKF (RCP high vibration). A self-checking circuit provides a green light when the monitor is operable. An analog output jack is available at the back of the monitor as well as output jacks for frequency filters, oscilloscopes and other devices on the face of the panel.

A sixteen point recorder (0 to 30 mils) located below the monitoring panel records all shaft and frame outputs, while two trend recorders can be selected to any RCP to record it's frame and shaft vibration trends.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 51	Group #	1	
	K/A #	073K1.01	
OK	Importance Rating	3.6	

Knowledge of PRM system design features and / or interlocks which provide for release termination when radiation exceeds setpoints

Proposed Question: Common 46

Given the following plant conditions:

- A plant heatup is in progress in Mode 3 at 450°F.
- A steam generator tube leak developed.
- SG activity increases to the SG blowdown isolation setpoint for R-19, S/G Blowdown Liquid Activity Monitor.
- SG blowdown isolates.
- Chemistry is requested to verify the Steam Generator activity level.

What is required to allow the Chemist to sample the Steam Generators for activity?

A. Remove the high radiation close signal by pulling the R-19 fuses at Bantam 11 cabinet.

B. Place the SG Blowdown Sample Isolation Valve switches in the PAB Sample Room to the Rad BYP position for the SG to be sampled.

C. Hold the SG Blowdown Isolation Valve control switch on panel SCF in the open position using a device made for this purpose for the SG to be sampled.

D. Only by raising the R-19 alarm setpoint to a value which is above the leaking SG activity.

Proposed Answer:

B. Place the SG Blowdown Sample Isolation Valve switches in the PAB Sample Room to the Rad BYP position for the SG to be sampled.

Explanation (Optional):

Technical Reference(s): SD-7
3-SOP-SG-1, ARP-040 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-MFW002 0005 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: 10/1/2002 Diablo Canyon 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

a SG tube leak is high activity in the blowdown fluid, as indicated by the R-19 blowdown monitor or by a grab sample. R-19 is common to all SG blowdown lines. If more than one SG is providing blowdown sample flow, then R-19 does not indicate which SG has the tube leak. If R-19 reaches its alarm setpoint, the following occurs:

- Blowdown sample containment isolation valves close,
- Blowdown containment isolation valves close, and
- City water to blowdown tank isolation valves closes.

Note that closing the blowdown sample containment isolation valves stops flow to the blowdown sampling system and to R-19. If a SG sample is necessary the containment isolation valve control switch is placed in RAD BYPASS. Sampling may be required to:

- Identify the affected SG
- Quantify the magnitude of the leak.

If blowdown is not isolated automatically by an R-19 alarm, the operators isolate blowdown from the leaking SG. If the leaking SG has not been identified, blowdown is secured from all SG. Isolating blowdown reduces the radiological release and reduces the amount of contamination in the blowdown purification system.

The remainder of the actions taken in the SG Tube Leak procedure involve:

- Determining whether the magnitude of the leak requires unit shutdown—for example, leak is in excess of the Technical Specification limits for primary to secondary SG leakage).
- Determining Emergency Classification based on leak rate
- Determining leak rate and shutdown criteria.
- Isolating the leaking SG to the greatest degree possible to minimize the radiological release.
- If and when the turbine is shutdown, the leaking SG is isolated completely (shut MSIV and bypass).
- If and when the reactor is shutdown, the RCS temperature and pressure are reduced to:
  - To block SI
  - RCS pressure is further reduced until it is equalized with SG pressure equals reduce SG leakage
  - Maintain an adequate minimum subcooling margin.

INPUT DEVICE: 3RM019

R19 S.G. BLDWN

SETPOINT: Variable

1.0 CAUSES

1.1 Steam generator blowdown high activity (SG tube leak)

2.0 AUTOMATIC ACTIONS

2.1 SG blowdown and sample valves close.

3.0 SUBSEQUENT ACTIONS

3.1 IF a Steam Generator tube leak is suspected,
THEN GO TO 3-AOP-SG-1, Steam Generator Tube Leak.

3.2 GO TO ONOP-RM-2, High Activity - Radiation Monitoring System.

4.0 REFERENCES

4.1 3-AOP-SG-1, Steam Generator Tube Leak.

4.2 ONOP-RM-2, High Activity - Radiation Monitoring System.

CAUTION

The Rad Bypass position SHALL NOT be used if the unit is in Modes 1 OR 2.

NOTE

In accordance with Con Edison - Memorandum Of Understanding No. 15, the Plant Manager (currently titled General Manager Operations) of IP3 SHALL authorize use of emergency surge waste space (i.e., SGBD capacity) at Con Edison's Unit No. 1 waste collection tanks.

4.5 Establishing SGBD to Unit 1

____ 4.5.1 COMPLETE SGBD initial lineup per Step 4.1 for SGs selected for blowdown to Unit 1.

____ 4.5.2 REQUEST Unit 2 Shift Manager to verify the following:

- ____ P-MT-122, Pressure Test of Secondary Boiler Blowdown Purification System, has been performed.
- ____ Unit 3 to Unit 1 Secondary Boiler Blowdown Purification System (SBBPS) is available.

____ 4.5.3 PLACE Containment Isolation valves control switches for affected SG(s) in RAD. BYPASS.

31 SG:

- ____ PCV-1214: Blowdown Isol Vlv 1 31 Steam Generator
- ____ PCV-1214A: Blowdown Isol Vlv 2 31 Steam Generator

32 SG:

- ____ PCV-1215: Blowdown Isol Vlv 1 32 Steam Generator
- ____ PCV-1215A: Blowdown Isol Vlv 2 32 Steam Generator

33 SG:

- ____ PCV-1216: Blowdown Isol Vlv 1 33 Steam Generator
- ____ PCV-1216A: Blowdown Isol Vlv 2 33 Steam Generator

34 SG:

- ____ PCV-1217: Blowdown Isol Vlv 1 34 Steam Generator
- ____ PCV-1217A: Blowdown Isol Vlv 2 34 Steam Generator

____ 4.5.4 OPEN BD-23, Steam Generator Blowdown To Unit 1 Header Isolation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 52	Group #	1	
	K/A #	076K2.01	
OK	Importance Rating	2.7	

Knowledge of bus power supplies to the service water

Proposed Question: Common 47

Service Water Pumps 34, 35 and 36 are lined up to the essential header, their Zurn Strainers are directly energized from ...

A. 5A, 2A and 6A

B. 5A, 3A and 6A

C. MCC 312A

D. MCC 36A and 36B

Proposed Answer:
D. MCC 36A and 36B

Explanation (Optional):

Technical Reference(s): 3-COL-EL-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-SW001 0004 (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 5

55.43 5

Comments:

LOS 52

	Des. Oper Pos.	Actual Pos.	Initial	Verifier

3.2 PAB - 55' Elev.

3.2.1 MCC-36A

	Des. Oper Pos.	Actual Pos.	Initial	Verifier
31 Electrical Tunnel Exhaust Fan	On			
Charging Pumps Seal Leakage Collection Tank Pumps A & B	Off			
31 33 35 SW Pump Strainers & Throw-Over Switch for 37 38 39 SW Pump Strainers	On			
RC-MOV-536 Pressurizer PORV PCV-456 Isolation	On			
AC-MOV-822A 31 RHR Heat Exchanger CCW Outlet Isolation	On			
AC-MOV-745B 32 RHR Heat Exchanger Inlet Isolation	On			
SI-MOV-746 32 RHR Hx Outlet Injection Isolation	On			
AC-MOV-730 RHR Loop Suction Isolation	Off			
AC-MOV-797 RCP CCW Supply Isolation	On			
AC-MOV-784 RCP CCW Bearing Return Isolation	On			
AC-FCV-625 RCP CCW Thermal Barrier Return Isolation	On			
AC-MOV-744 RHR Pumps Discharge Isolation	Locked Off			
SI-MOV-866A 31 Spray Pump Discharge Isolation	On			
SI-MOV-856G Loop 31 Hot Leg High Head Injection Line BIT Header Isolation	Locked Off			
SI-MOV-880A 31 FCU Charcoal Filter Dousing Isolation	On			
SI-MOV-880C 32 FCU Charcoal Filter Dousing Isolation	On			
SI-MOV-880E 33 FCU Charcoal Filter Dousing Isolation	On			

	Des. Oper Pos.	Actual Pos.	Initial	Verifier
BFD-MOV-2-31 31 Main Boiler Feed Pump Discharge Isolation	On	______	______	______
SI-MOV-856E Loop 31 Cold Leg High Head Injection Line BIT Header Isolation	On	______	______	______
SI-HCV-640 32 RHR Hx Outlet Flow Control Valve	On	______	______	______
Spare	Off	______	______	______
33 Control Building Exhaust Fan	On	______	______	______
Spare	Off	______	______	______
Spare	Off	______	______	______
Spare	Off	______	______	______
Spare	Off	______	______	______
Spare	Off	______	______	______
Spare	Off	______	______	______

3.2.2 MCC-36B

	Des. Oper Pos.	Actual Pos.	Initial	Verifier
32 Electrical Tunnel Exhaust Fan	On	______	______	______
Backup Feed to Instrument Busses 34 & 34A	On	______	______	______
32 34 36 SW Pump Strainers & Throw-Over Switch for 37 38 39 SW Pump Strainers	On	______	______	______
RC-MOV-535 Pressurizer PORV PCV-455C Isolation	On			
AC-MOV-822B 32 RHR Heat Exchanger CCW Outlet Isolation	On	______	______	______
AC-MOV-745A 32 RHR Heat Exchanger Inlet Isolation	On	______	______	______
SI-MOV-899A 32 RHR Hx Outlet Injection Isolation	On	______	______	______
AC-MOV-731 RHR Loop Suction Isolation	Off	______	______	______
AC-MOV-769 RCP CCW Supply Isolation	On	______	______	______
AC-MOV-786 RCP CCW Bearing Return Isolation	On	______	______	______

b) Annunciated by:

(1) High strainer ΔP:

(a) > 7 psid for 30 seconds

(2) Loss of power to control circuit:

(a) Strainer motor overload

(b) Blown fuses to control circuit

(c) Loss of 480VAC to strainer motor

2) BACKUP SERV. WATER STRAINERS TROUBLE:

a) CR panel SJF

b) Same conditions as main SWS strainer trouble alarm

3) If ΔP reaches 9 psid, System Engineering evaluation required — 3-SOP-RW-005 4.4.2.2

f. Control Panels, ΔP sensing lines, and gauges are heat traced for freeze protection

g. Powered Supplies: — EO 0004

1) 31, 33 and 35:

a) MCC-36A

2) 32, 34 and 36:

a) MCC-36B

3) 37, 38 and 39:

a) Either MCC-36A or MCC-36B

b) Throw switch in 480V switchgear room

h. Instrumentation: — EO 0002

1) 0-10 psid indication for each Zurn strainer — MOD 91-3-128

a) Normal ΔP = 4 psid

2) Amber light - overload

3) Zurn Strainer Pit Temperature Monitor: — MOD 93-3-376

a) Alarms at 45°F

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 53	Group #	1	
	K/A #	078G2.4.4	
OK	Importance Rating	4.0	

Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: Common 48

Given the following:

- 75% power
- PRZR Pressure 2265 slowly increasing
- PRZR spray valves closed
- Letdown Orifice Valves closed
- Charging flow increasing
- PRZR Level 54% and increasing

Which one of the following malfunctions would cause these indications?

A. Controlling Pressurizer Pressure Channel Failed Low

B. Controlling Pressurizer Level Failed Low

C. Pressurizer Pressure Master Controller Failure

D. Loss of Instrument Air

Proposed Answer:
D. Loss of Instrument Air

Explanation (Optional):

Technical Reference(s): 3-AOP-AIR-1
SD-3.0
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-IA001 7 (1799) (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

WS 53

Attachment 1
Valves of Immediate Concern

CHEMICAL AND VOLUME CONTROL SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
CH-LCV-459	LETDOWN ISOLATION VALVE	Closed
CH-LCV-460	LETDOWN ISOLATION VALVE	Closed
CH-AOV-200A	LETDOWN CONTROL VALVE	Closed
CH-AOV-200B	LETDOWN CONTROL VALVE	Closed
CH-AOV-200C	LETDOWN CONTROL VALVE	Closed
CH-AOV-201	LETDOWN ISOLATION VALVE	Closed
CH-AOV-202	LETDOWN ISOLATION VALVE	Closed
CH-FCV-110A	BORIC ACID BLENDER BORIC ACID FLOW CONTROL VALVE	Open
CH-FCV-110B	VCT MAKEUP VALVE	Closed
CH-FCV-111A	MAKEUP H2O TO BORIC ACID BLENDER	Closed
CH-FCV-111B	VCT MAKEUP VALVE	Closed
CH-HCV-142	CHG. LINE FLOW CONTROL VALVE	Closed
CH-AOV-261A	31 SEAL RETURN ISOLATION VALVE	Open
CH-AOV-261B	32 SEAL RETURN ISOLATION VALVE	Open
CH-AOV-261C	33 SEAL RETURN ISOLATION VALVE	Open
CH-AOV-261D	34 SEAL RETURN ISOLATION VALVE	Open
CH-AOV-246	RCP'S NO.1 SEAL BYPASS	Closed
CH-HCV-123	EXCESS LTDN FLOW CONTROL	Closed
CH-HCV-133	RESID HR LP BYPASS TO DEMIN	Closed
CH-AOV-204A	LOOP 32 ALTERNATE CHARGING ISOLATION	Open
CH-AOV-204B	LOOP 31 NORMAL CHARGING ISOLATION	Open
CH-TCV-100	BATCHING TANK AUXILIARY STEAM TEMPERATURE CONTROL VALVE	Open

Attachment 1
Valves of Immediate Concern

REACTOR COOLANT SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
RC-PCV-455A	PRZR SPRAY VLV LOOP 34	Closed
RC-PCV-455B	PRZR SPRAY VLV LOOP 33	Closed
NNE-AOV-863	ACCUMULATOR N2 SUPPLY VALVE	Closed
RC-AOV-519	PRIMARY WATER CONTAINMENT ISOLATION VALVE	Closed
RC-AOV-552	PRIMARY WATER CONTAINMENT ISOLATION VALVE	Closed
RC-AOV-548	PRESSURIZER RELIEF TANK TO GAS ANALYZER ISOLATION	Closed
RC-AOV-549	PRESSURIZER RELIEF TANK TO GAS ANALYZER ISOLATION	Closed
RC-AOV-550	N2 TO PRT	Closed

SAFETY INJECTION SYSTEM		
VALVE	**FUNCTION**	**FAIL POSITION**
SI-AOV-1851A SI-AOV-1851B	BIT RECIRCULATION VALVES	Closed
SI-AOV-876A SI-AOV-876B	SPRAY ADDITIVE TANK DISCHARGE ISOLATION	Open

from the alternate source, only local operation of the pump breaker at MCC-312A is available. Normal speed control is available if instrument buses are energized.

Charging pump speed and flow rate is automatically controlled as a function of pressurizer level. The pressurizer level programmer (TC-412N in control room rack D-8) produces an output linearly proportional to T_{AVG} therefore a constant RCS mass is maintained during load changes and steady state operation. This output is compared, by LC459D in rack B-6, to actual pressurizer level (see graph RCS-10) and the resulting output represents the error signal. The output is transmitted from the master level controller to the three charging pump speed controllers located on flight panel FBF. When the master auto-manual controller is in the AUTO position, it passes the pressurizer level error signal to the three pump speed controllers and controls which ever pump is in AUTO. When the master auto-manual controller is in MANUAL, the pressurizer error signal is blocked and the operator can control the pump speed by adjusting the master auto-manual controller. The output of the individual pump controller is used to drive the positioner to properly position the scoop tube in the fluid drive coupling. The level programmer (TC-412N) is a proportional controller (reset is dialed out), the master controller (LC-459D) is a proportional-integral-derivative controller (derivative is set to zero), and the charging pump speed controllers are proportional controllers. Each pump has an associated run time meter which displays (run time in hours) inside the flight panel.

Each charging pump has an adjustable low speed stop on the scoop tube to ensure the reactor coolant pumps have seal water flow at all times. This stop is adjustable to allow a higher flow to be set if mechanical seal leakage increases. If the electrical signal from the control room speed controller is lost, the pumps will go to minimum speed. When this occurs, the pump speed can be controlled from the local panel on the 55 foot PAB elevation by placing the controllers in MANUAL. If instrument air is lost due to a slow leak, 31, 32 and 33 charging pumps will fail to maximum speed. If the loss of instrument air is a sudden catastrophic loss the simultaneous loss of pressure on the upper and lower chambers of the piston will cause the actuator to fail "as is". In this situation, pump speed can be controlled by isolating and venting the air to the scoop tube positioner and manually positioning it. This is listed as a subsequent action in ONOP-IA-1, Loss of Instrument Air.

A "Low Charging Flow" alarm on CVCS supervisory panel SFF is derived from the magnitude of the signal to the pump speed controllers. If the pressurizer level is above its setpoint for an

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 54	Group #	1	
	K/A #	103A3.01	
OK	Importance Rating	3.9	

Ability to monitor automatic operation of the containment system including containment isolation

Proposed Question: Common 49

Given the following conditions:

- A Large Break LOCA has occurred
- Containment pressure is 27 psig
- Containment Spray failed to AUTO actuate
- Manual Containment Spray actuation was successful

What affect does this condition have on Containment Phase B isolation?

A. Phase B will automatically actuate by the Manual Containment Spray signal.

B. Phase B must be manually actuated using 1 of 2 pushbuttons.

C. Phase B must be manually actuated using 2 of 2 pushbuttons.

D. The Phase B valves must be manually closed using individual control switches in the control room.

Proposed Answer:

A. Phase B will automatically actuate by the Manual Containment Spray signal.

Explanation (Optional):

Technical Reference(s): SD-10.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-SIS001 C (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS 54

		(Second Opportunity
32 CS Pump	Immediately	With Containment Spray and No SI signal (SI reset)

2.3.10 Containment Spray (Figure 10.0-6 & 7)

The Containment Spray system reduces containment pressure and removes fission products from containment atmosphere. Pressure is reduced by condensing steam present inside the containment. Fission products (particularly I-131) are removed by absorption in the spray water. Sodium Hydroxide is added to the spray water to create a more soluble form of Iodine in the containment and recirculation sumps.

Containment pressure is monitored by 6 pressure detectors arranged into two groups (948A, 948B, & 948C and 949A, 949B, & 949C). When two of three pressure channels in both groups reach 22 psig, an automatic Containment Spray Actuation occurs.

Manual Containment Spray is actuated using 2 of 2 pushbuttons. This electrical arrangement prevents inadvertent actuation of Containment Spray.

An Automatic or Manual Containment Spray Signal will cause:

- Actuation of Containment Spray
- Containment Phase B Isolation
- Containment Ventilation Isolation

As with other actuation signals, it may be necessary to manually control components while containment conditions still require a containment spray (i.e., pressure > 22 psig). The containment spray signal can be reset by depressing train specific reset pushbuttons. The containment spray signal is "blocked" and components that receive containment spray signals can be manually operated. If containment pressure decreases to less than 22 psig, and the containment spray actuation signal is removed, and the re-initiation "block" is removed. This means that if containment pressure subsequently increases to greater than 22 psig, containment spray will re-actuate.

A Containment Spray actuation signal will:

- Send an open signal to spray pump motor operated discharge valves 866A and 866B, and
- Start the 31 (32) Containment Spray Pumps under one of the three following conditions:

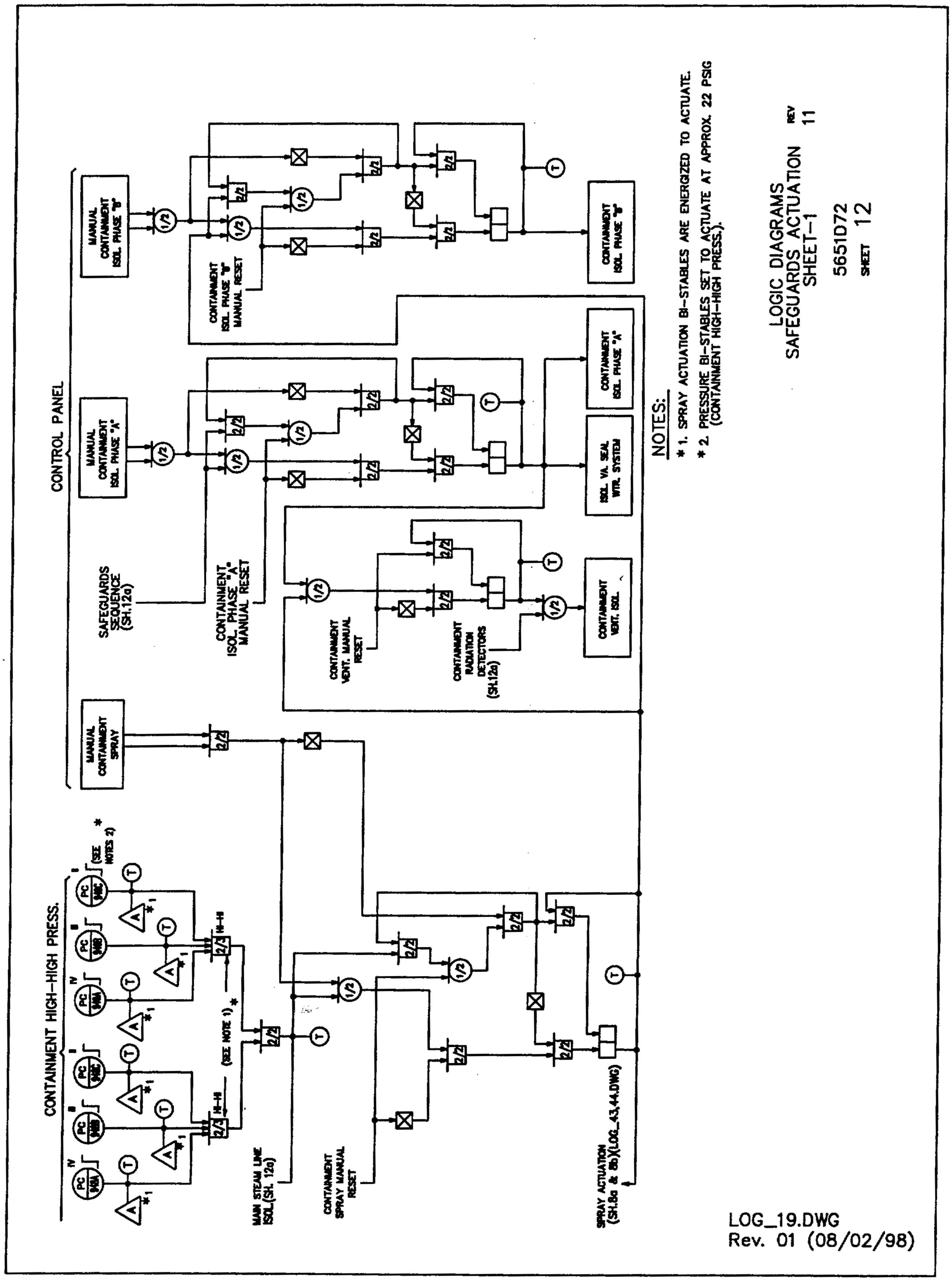


Figure 10.0-6: Vapor Containment Pressure Sensing for Train "A" (LOG_19.DWG)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 50	Group #	1	
	K/A #	064K2.03	
OK	Importance Rating	3.2	

Knowledge of bus power supplies to the control power

Proposed Question: Common 50

An electrical short caused a loss of 33 DC Power Panel. Subsequently an inadvertent Safety Injection signal was generated. What would be the configuration of the Emergency Diesel Generators following the SI?

A. ALL three EDGs would be running.

B. Only 31 and 32 EDGs would be running.

C. Only 32 and 33 EDGs would be running.

D. Only 31 and 33 EDGs would be running.

Proposed Answer:
C. Only 32 and 33 EDGs would be running.

Explanation (Optional):

Technical Reference(s): 3-AOP-DC-1, Attachment 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EDSEDG 3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

WS 50

Attachment 9
Loads on 33 DC Power Panel

Page 1 of 1

Ckt	Load	Comment
1	480 V SWGR 31 Bus 2A Bkr Cont. & Bus 2A, 3A Safeguards	
2	480 V SWGR 32 Bus 3A Bkr Control	
3	33 - 34 DC Power Panel Tie Breaker	
4	Diesel Gen 31 Control Panel	
5	Supervisory Panel SC - Aux. BFP 31 Recirc Valve (SOV-1321)	
6	34 Cont. Recirc Fan SOV	
7	32 Cont. Recirc Fan SOV	
8	Supervisory Panel SB-1 - Independent Indication System	
9	Service Water Mode Selector Switch	
10	Spare	
11	Spare	
12	Spare	

••• END •••

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 55	Group #	1	
	K/A #	103A4.04	
OK	Importance Rating	3.5	

Ability to manually operate and / or monitor in the control room Phase A and Phase B resets

Proposed Question: Common 51

The plant was operating in the normal full power lineup. An inadvertent Safety Injection and Phase A Isolation occurred. Safety Injection has been RESET. What action will allow Letdown Isolation Valves, 201 and 202, to be opened?

A. Depress both Phase A master Reset pushbuttons and the valves will reopen.

B. Depress both Phase A master Reset pushbuttons then put 201 and 202 valves switches to close and then back to open and the valves will reopen.

C. Depress both individual Reset pushbuttons then put 201 and 202 valves switches to close and then back to open and the valves will reopen.

D. Depress both Phase A master Reset pushbuttons then depress the individual valve Reset pushbuttons and the valves will reopen.

Proposed Answer:

D. Depress both Phase A master Reset pushbuttons then depress the individual valve Reset pushbuttons and the valves will reopen.

Explanation (Optional):

Technical Reference(s): 3-ES-1.1, Attachment 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CVC001 5.0 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

ATTACHMENT 3 (Attachment page 14 of 19)

RE-ESTABLISHING OPERATOR CONTROL OF VALVES FOLLOWING PHASE A RESET

(SECTION II continued from previous page)

b. Panel SNF

Reset Button	Associated Valve(s)
1	CA-PCV-1230 - SJAE to V.C.
2	CA-PCV-1229 - SJAE to V.C.
3	AC-AOV-798 - CCW Inlet to Excess. Ltdn. Hx
4	AC-AOV-791 - CCW Inlet to Excess. Ltdn. Hx
5	AC-AOV-796 - CCW Outlet from Excess Ltdn. Hx
6	AC-AOV-793 - CCW Outlet from Excess Ltdn. Hx
7	RC-AOV-548 - PRT to Gas Analyzer
8	RC-AOV-519 - PW Cont. Isol.
9	VS-PCV-1235 & 1237 - V.C. Rad. Monitor PS-PCV-1239 & 1241 - WCCPP to VC Rad Monitor
10	VS-PCV-1234 & 1236 - V.C. Rad. Monitor PS-PCV-1238 & 1240 - WCCPP to VC Rad Monitor
11	CH-AOV-202 - Letdown
12	CH-AOV-201 - Letdown
31	RC-AOV-552 - PW Cont. Isol.
33	RC-AOV-549 - PRT to Gas Analyzer

(SECTION II CONTINUED ON NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 56	Group #	2	
	K/A #	001K3.02	
OK	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the CRDS will have on the RCS

Proposed Question: Common 52

The following conditions exist on Unit 3:

- The plant was operating at 100% power when a Main Turbine Control Oil leak caused a very slow turbine load rejection to 90% power
- Prior to and during the load rejection Rod Control was in Manual
- Tave increased from 567°F to 575°F
- The CRS orders Rod Control be placed in MANUAL

When Rod Control is placed from MANUAL to AUTO, Control Rod Speed should go from ______ steps per minute (SPM) in MANUAL to ______ SPM in AUTO?

A. 72 SPM in MANUAL, 66 SPM in AUTO

B. 66 SPM in MANUAL, 72 SPM in AUTO

C. 72 SPM in MANUAL, 8 SPM in AUTO

D. 66 SPM in MANUAL, 8 SPM in AUTO

Proposed Answer:

B. 66 SPM in MANUAL, 72 SPM in AUTO

Explanation (Optional):

Technical Reference(s): SD-16.1 page 28 and Figure 16.1-28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICROD G (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: 7/17/2002 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 57	Group #	2	
	K/A #	002K6.02	
OK	Importance Rating	3.6	

Knowledge of the effect or a loss or malfunction of the RCP on the RCS

Proposed Question: Common 53

Unit 3 was operating at 28% power when 31 Reactor Coolant Pump (RCP) tripped on overcurrent.

Which of the following describes the unit's initial response? (Assume NO operator action AND NO rod motion.)

A. A reactor trip occurs and unaffected loops T_{AVE} decreases.

B. A reactor trip occurs and unaffected loops T_{AVE} increases.

C. A reactor trip will NOT occur and unaffected loops T_{AVE} decreases.

D. A reactor trip will NOT occur and unaffected loops T_{AVE} increases.

Proposed Answer:

D. A reactor trip will NOT occur and unaffected loops T_{AVE} increases.

Explanation (Optional):

Technical Reference(s): SD-28
3-ARP-002 page 4
Simulator

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICRXP E-3 (As available)

Question Source: Bank # INPO 20589
Modified Bank # (Note changes or attach parent)
New

Question History: 6/29/2000 Byron 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

WS 57

LOSS OF
FLOW
SINGLE LOOP

INPUT DEVICE:	Loop Flow	RCP Breakers	
	FC-414	52b	RCP-31x
	FC-415 LF-1X	RCP31	
	FC-416		
	FC-424	52b	RCP-32x
	FC-425 LF-2X	RCP32	
	FC-426		
	FC-434	52b	RCP-33x
	FC-435 LF-3X	RCP33	
	FC-436		
	FC-444	52b	RCP-34x
	FC-445 LF-4X	RCP34	
	FC-446		

SETPOINT: 93% of full flow
RCP breaker open

NOTE

This trip and alarm are bypassed below P-8 permissive (approximately 35% power).

1.0 CAUSES

1.1 Low flow in any 1 of 4 loops as indicated by 2 out of 3 flow transmitters in loop when power is greater than P-8 setpoint.

1.2 Any Reactor Coolant Pump breaker open when power is greater than P-8 setpoint.

2.0 AUTOMATIC ACTIONS

2.1 If power is greater than P-8 the reactor trips.

(CONTINUED ON THE NEXT PAGE)

2.2.2 P-7 Permissive (Figure 28-15)

The P-7 permissive is used to block the high pressurizer level, low pressurizer pressure reactor trips, reactor coolant low flow and undervoltage reactor trip signals to the reactor protection system. The P-7 permissive is activated by a bistable circuit indicating less than 10% power as measured by both turbine first stage pressure detectors and 3/4 power range channels. The power range input is supplied by the P-10 permissive. A white "POWER BELOW P-7" lamp illuminates on the control room FBF panel while the P-7 permissive is active and extinguishes when reactor power and/or turbine power are >10%.

2.2.3 P-8 Permissive (Figure 28-14)

The P-8 permissive blocks the automatic reactor trip on low flow in one loop if power is below 35% at the time one RCP is lost. The permissive also blocks a reactor trip due to a turbine trip when power is below 35%. A white "POWER BELOW P-8" lamp illuminates on the control room FBF panel when the P-8 permissive is active and extinguishes when reactor power is above 35%.

2.2.4 P-10 Permissive (Figure 28-14)

The P-10 permissive blocks the intermediate range channel and low power range channel trips during an approach to power. It is also used to backup the P-6 permissive to block the Source Range instrumentation and is one of the inputs to the P-7 permissive.

When 2 of 4 power range channels indicate greater than 8.5% power the P-10 permissive is activated and a white "POWER ABOVE P-10" lamp illuminates. Once the P-10 lamp is lit, the low power and intermediate range hi flux trips may be manually blocked as described in the sections for those trips.

The P-10 permissive and associated manual blocks are automatically reinstated if power falls below 8.5% on 3/4 Power Range channels.

2.2.5 Low Power Auto Rod Withdrawal Block

Automatic control rod withdrawal is blocked until turbine power, as sensed by PT-412A (turbine first stage pressure), is greater than 15%. Automatic control rod insertion is not blocked. A white "LOW PWR AUTO ROD WITHDRAWAL BLOCK" light, on the FBF panel, is illuminated when the permissive is active and extinguishes when turbine power is >15%.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 48	Group #	1	
	K/A #	063A3.01	
OK	Importance Rating	2.7	

Ability to monitor automatic operation of the DC electrical system meters, annunciators, dials, recorders and indicating lights

Proposed Question: Common 54

Given the following plant conditions:

- A BATTERY CHARGE TROUBLE category alarm was received in the control room
- The conventional NPO reports the RED light for + (positive) Ground Detection for 31 Battery Charger is LIT

What actions are required (if any) to clear the alarm in the control room making it available to alarm on any future alarm condition?

A. Push the Acknowledge and Reset pushbuttons in the control room.

B. Place the Normal/Bypass switch on 31 Battery Charger in Bypass position.

C. Open 31 Battery Charger output breaker.

D. Cannot be cleared until the + ground condition is corrected.

Proposed Answer:

B. Place the Normal/Bypass switch on 31 Battery Charger in Bypass position.

Explanation (Optional):

Technical Reference(s): SD-27.5 (Attach if not previously

provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EDS125 E-7 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

clear the input to the common alarm on supervisory panel SHF, and reset the HV S/D circuit for the battery charger.

FLOAT ADJUST--screw type adjust potentiometer used to adjust the nominal float voltage that the charger will maintain on the battery.

EQUALIZE ADJUST--screw type adjust potentiometer used to adjust the equalizing voltage used when performing an equalizing charge on the battery.

EQUALIZE/FLOAT SELECT TOGGLE SWITCH--this toggle switch has two positions, equalize and float, and is used to select which type of charge is being performed on the battery.

ALARM BYPASS TOGGLE SWITCHES--Six toggle switches are provided under a plexiglass cover on the charger control panel to bypass individual alarm functions in the event of a spurious alarm circuit or for maintenance. The switches have two positions, normal and bypass. In NORMAL, each of the alarm functions input to the common "BATTERY CHARGER TROUBLE" alarm on supervisory panel SHF. In BYPASS, the inputs to the common alarm are disabled and an amber light above the switch lights. This switch will not affect the local alarm buzzer on the charger control panel. The following alarms are inputs to the common alarm on SHF from 31 and 32 battery chargers:

NOTE: The local alarms for 35 battery charger are *not* wired into the supervisory alarm on SHF

- LOW DC VOLTAGE
- GROUNDS POS AND NEG
- AC FAILURE
- BATTERY DISCHARGE
- OVERTEMPERATURE
- CHARGER SHUTDOWN

COMMON ALARM BUZZER ON/OFF TOGGLE SWITCH--this switch is used to turn on and off the common alarm buzzer at the battery charger. An amber light above the switch will light when the switch is in the off position. This switch disables the local buzzer for the following alarms:

- LOW DC VOLTAGE
- GROUNDS POS AND NEG
- AC FAILURE

- BATTERY DISCHARGE
- OVERTEMPERATURE

AC VOLTAGE SELECTOR SWITCH--this switch is used to select which voltage will be read on the AC voltmeter. It is a four position switch, L1-L2, L2-L3, L3-L1 and OFF.

AC AMPERAGE SELECTOR SWITCH--this switch is used to select which of the phases will be read on the AC ammeter. It's a (3) position switch, 1, 2 and 3.

2.1.3.3 Indications 31, 32, and 35 Chargers (Figure 27.5-4)

AC ON--a green light on the control panel indicating that AC power is available at the charger (i.e. the AC input breaker is closed and power is available from the MCC).

AC FAILURE--a red light indicating that AC power to the charger has been lost.

DC VOLTAGE--a 0-200 volt meter that indicates the DC voltage out of the charger.

DC AMPS--a 0-500 amp meter that indicates the DC load on the battery charger.

GROUND DETECTION--Two red lights are provided for ground detection, 1 for (+) ground and 1 for (-) ground which monitor the (+) and (-) terminal voltage to chassis ground. With no ground present, both lights are de-energized. If a ground condition occurs, the light will illuminate. Once the relay has been energized and the light lit for this type of ground detection circuit, the relay must be reset. This is done with the ground detection relay reset switch.

HIGH VOLTAGE SHUTDOWN--this is a red light on the control panel that will light if the charger has been shutdown due to high voltage. The setting is 146 volts DC. Once the charger is shutdown on high voltage the reset toggle is used to reset the charger to operation and clear the local and common category alarm on SHF. This alarm input to the common battery trouble alarm on SHF cannot be bypassed.

LOW VOLTAGE--a red light on the control panel to indicate that the DC output of the charger is below 116 volts .

OVERTEMP--a red light on the control panel to indicate an over temperature condition exists i.e., >160°F.

BATTERY DISCHARGE--an amber light on the control panel to indicate that the battery is no longer receiving charging current. This is done by monitoring the direction of current flow in a DC

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 58	Group #	2	
	K/A #	011K5.15	
OK	Importance Rating	3.6	

Knowledge of the operational implications of PZR level indication when RCS is saturated as it applies to the PZR LCS

Proposed Question: Common 55

Given the following conditions:

- A Loss of Offsite Power has occurred.
- RCS cooldown is being performed in accordance with ES-0.2, Natural Circulation Cooldown.
- RCP's cannot yet be started.
- The RO is depressurizing using auxiliary spray.
- PRZR level rapidly rises from 24% to 66%.

Which one of the following describes the reason for the Pressurizer level increase?

A. Loss of Secondary Heat Sink.

B. Charging flow is refilling the Pressurizer as RCS pressure drops.

C. Cooldown rate is not high enough to maintain Pressurizer level with auxiliary spray in service.

D. Portions of the RCS have reached saturation temperature.

Proposed Answer:

D. Portions of the RCS have reached saturation temperature.

Explanation (Optional):

Technical Reference(s): ES-0.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPE00 G (As available)

Question Source: Bank # INPO 24956
Modified Bank # (Note changes or attach parent)
New

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

ws 58

Number:	Title:	Revision Number:
ES-0.2	NATURAL CIRCULATION COOLDOWN	17

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	<u>CONTINUE RCS Cooldown And Depressurization:</u>	
	a. MAINTAIN cooldown rate in RCS cold legs - LESS THAN 25°F/HR	
	b. MAINTAIN RCS subcooling based on qualified core exit TCs - GREATER THAN 90°F	b. STOP depressurization and REESTABLISH subcooling.
	c. MAINTAIN RCS cold leg temperature and pressure - WITHIN LIMITS OF OPERATIONS GRAPH RCS-1B	
15.	<u>VERIFY Reactor Vessel - FREE OF STEAM VOIDS</u> • PRZR Level - NO UNEXPECTED LARGE VARIATIONS • RVLIS Full Range - GREATER THAN 100%	PERFORM the following: a. Re-PRESSURIZE RCS within the limits of Operations Graph RCS-1B to collapse potential voids and CONTINUE cooldown. b. <u>IF</u> RCS depressurization must continue, <u>THEN</u> GO To ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 59	Group #	2	
	K/A #	014K1.02	
OK	Importance Rating	3.0	

Knowledge of the physical connections and / or cause effect relationships between the RPIS and the NIS

Proposed Question: Common 56

The following conditions exist on Unit 3:

- Reactor power is holding steady at 1E-8 amps during a normal reactor startup
- Individual and group position indicators show all control bank D rods at 120 steps withdrawn

When the ATC begins to withdraw control rods to raise reactor power, both IR NIS indications suddenly drops by 1/3 decade and continues to decrease at a negative (-) 0.25 DPM. There is no significant change in RCS T_{AVE}. The control bank D step counters now read 121 steps for groups 1 and 2. IRPI indicators for rods P-10, K-2 and H-8 (CBD Group 2) indicate 0 steps. All other rod position indicators (IRPIs) are unchanged.

Which of the following has occurred based on these indications?

A. The control bank D group 2 step counter has failed; it should also read 0 steps if the rods in this group are fully inserted.

B. The individual rod position indicators appear to have failed, more than a single dropped rod would have resulted in a reactor trip.

C. The control bank step counters and associated IRPI indicators, along with the NIS indications are consistent with multiple dropped rods.

D. Either the control bank D group step counter or 3 IRPI indicators have failed, not enough information is provided to determine which.

Proposed Answer:

C. The control bank step counters and associated IRPI indicators, along with the NIS indications are consistent with multiple dropped rods.

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICROD C (As available)

Question Source: Bank # INPO 21426
Modified Bank # (Note changes or attach parent)
New

Question History: 7/17/2002 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3,9
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 49	Group #	1	
	K/A #	063K3.02	
OK	Importance Rating	3.5	

Knowledge of the effects that a loss or malfunction of the DC electrical system will have on components using DC control power

Proposed Question: Common 57

Given the following:

- The plant is at 100% power.
- Safeguards Train B, DC Power has failed.

Which ONE of the following describes the response of AFW Pump 33 to a safeguards actuation?

A. Pump will auto start but does NOT supply water to any S/G.

B. Pump will auto start and supplies water to only 34 S/G.

C. Pump will auto start and supplies water to 33 and 34 S/Gs.

D. Pump will NOT auto start.

Proposed Answer:

D. Pump will NOT auto start.

Explanation (Optional):

Technical Reference(s): SD-10.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EDS125 E-5 (As available)

Question Source: Bank # INPO 28414
Modified Bank # (Note changes or attach parent)
New

Question History: 11/15/2004 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

2.4.5 Containment Spray Reset (Figure 10.0-6 & 7)

The Containment Spray reset signal is reset from CR panel SBF-1. Two black pushbuttons are provided, one for each safeguards logic channel. The spray signal can be reset with the Hi-Hi VC Pressure present, however, automatic Containment Spray reactuation will be prohibited until the High-High Containment Pressure signal has cleared. Manual actuation of the Containment Spray Signal can be initiated at any time.

2.5 Engineered Safeguards Power Supplies

Power for ESS equipment is derived either directly from the four 480 safeguards volt buses or via the safeguards motor control centers. These buses can be powered from one of two sources, either the 6.9 kV buses (via transformers) or from the emergency diesel generators.

The arrangement of the equipment on the 480 V buses ensures that minimum safeguards equipment are available with a single active failure. Accommodations for testing of components are also provided. Any loss of a single component or power supply still satisfies the minimum safeguards equipment.

Two logic circuits (Train A and Train B) exist for safeguards actuation. The logic relays associated with each actuation signal are powered from DC Distribution Panel 31 (Train A) and DC Distribution Panel 32 (Train B). The sequence signals associated with Train A and Train B provide signals in a manner that each would satisfy the minimum safeguards equipment. Train A feeds buses 5A, 2A, and 3A. Train B feeds buses 6A, 2A, and 3A. The Bus Sequencing relays are powered as follows: DC Power Panel 31: Bus 5A., DC Power Panel 33: Buses 2A and 3A., and DC Power Panel 32: Bus 6A. These circuits also serve as control power for the respective bus except Bus 3A (control power is from DC Power Panel 33 Circuit 2). Failure of control power is indicated by the alarm SAFE-GUARDS INITIATION RACK On 480 V SWGR. SEQ. DC POWER FAILURE on panel SBF-1.

2.5.1 SI With Off-Site Power Available

During a safety injection signal <u>with</u> offsite power available, the station auxiliary transformer, via bus 5 and 6 tie breakers, provides power to 6.9 kV buses 1-4. This tie is accomplished automatically on a unit trip when no bus faults are present and the breakers are not in the pullout position. Buses 2, 3, 5, and 6 provide power to 480 volt buses 2A, 3A, 5A, and 6A via Station Service Transformers. (Refer to System Description 27.4, Electrical Systems - Medium Voltage for details.)

While this is occurring, the safety injection signal strips the 480 volt buses of all non-required safeguards equipment (Non-Essential Load Shed). It also sends an open signal to the 480 volt bus tie breakers between buses 2A and 5A, and buses 3A and 6A. Motor Control Centers 36A - 36E remain energized during a Safety Injection. These MCCs service equipment needed by the Safeguards System. Refer to Figure 10.0-8 for EDG starting and 480V bus clearing logic diagram.

The safety injection signal starts the diesel generators in anticipation of a loss of off-site power. The SI signal also blocks the EDG overcurrent, and reverse power trips. This adds a margin of safety in case of a subsequent blackout by having power readily available.

The SI signal (with voltage available to the safeguards bus) energizes the following equipment:

Bus 5A	Bus 2A	Bus 3A	Bus 6A
31 SW Pump if selected	32 SW Pump if selected		33 SW Pump if selected
34 SW Pump if selected		35 SW Pump if selected	36 SW Pump if selected
		31 AFW Pump	33 AFW Pump
		32 AFW Pump	32 AFW Pump
31 FCU	32 FCU	34 FCU	35 FCU
33 FCU			
31 SI Pump	32 SI Pump		33 SI Pump
		31 RHR Pump	32 RHR Pump
31 CS Pump if spray signal			32 CS Pump if Spray signal

2.5.2 SI with a Loss of Off-site Power Condition

When a loss of offsite power to the 480 V buses occurs with a safety injection signal (SI Blackout), the following occurs:

- The 480 volt buses are isolated from the 6.9 kV buses by tripping breakers 52/2A, 52/3A, 52/5A and 52/6A. These bus supply breakers are tripped when an undervoltage condition is sensed on the respective bus. 480 volt tie breakers 2AT5A and 3AT6A (normally open and racked to test position) receive a trip signal to prevent fault transfer. The equipment on the 480 volt buses trips off due to the undervoltage, and all auto-starting non-safeguards equipment is locked out. The diesels start and come up to the idle condition. When the diesels are up to voltage and the buses are isolated, the breakers are closed

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 60	Group #	2	
	K/A #	015K5.02	
OK	Importance Rating	2.7	

Knowledge of the operational implications of discriminator/compensation operation concepts as they apply to the NIS

Proposed Question: Common 58

Which one of the following contains BOTH conditions that will result in indicated reactor power being LOWER than actual reactor power?

A. Source Range pulse height discrimination set too HIGH.
Intermediate Range compensating voltage set too HIGH.

B. Source Range pulse height discrimination set too LOW.
Intermediate Range compensating voltage set too LOW.

C. Source Range pulse height discrimination set too LOW.
Intermediate Range compensating voltage set too HIGH.

D. Source Range pulse height discrimination set too HIGH.
Intermediate Range compensating voltage set too LOW.

Proposed Answer:

A. Source Range pulse height discrimination set too HIGH.
Intermediate Range compensating voltage set too HIGH

Explanation (Optional):

Technical Reference(s): SD-13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICEXC E-3 (As available)

Question Source: Bank # INPO 24930

Modified Bank # (Note changes or attach parent)

New

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

attenuator selector switch, located inside the SR drawer assembly. The attenuator is preset to compensate for the input cable attenuation of the pulses.

After the pulses have been amplified, they are passed through a discriminator unit. The discriminator electrically gates or discriminates out pulses or heights below certain levels, so only those desired are read. The level below which the input pulses are discriminated is determined by adjusting the discriminator adjust potentiometer located inside the SR drawer. In this manner, the pulses caused by gamma events and electrical noise are removed. This produces a noise-free signal that can be accurately used to determine reactor neutron flux while in the SR.

Figure 13-13 shows the effect of changing the discriminator setting. At the minimum setting, the discriminator is not effective and the pulse amplifier will pass every event, gamma and neutron, onto the pulse counter. As the discriminator setting is increased, the pulse amplifier begins to ignore the lower energy, gamma events. As the lower graph shows, there is a gap between the energy level of a gamma event and the energy level of a neutron event. When the discriminator setting corresponds to this energy gap, the amplifier will pass all neutron events to the pulse counter, while ignoring all gamma events. If the discriminator is set too high, the pulse amplifier will begin to ignore the less energetic neutrons until finally the maximum setting will cause the amplifier to ignore all events regardless of energy level. Discrimination is important since the gamma rays from fission products, which accumulate in the fuel as reactor operations progresses, would mask the low neutron field during the initial phases of startup.

The pulse amplifier also provides an isolated output to drive audio circuits for generating an audible signal proportional to the count rate. A test-calibrate module, similar to that of the preamplifier, is provided to insert signals of 60, 10^3, 10^5 and 10^6 pulses per second into the pulse amplifier. Switching operations are controlled from the front of the Source Range drawer assembly.

2.6.4 Pulse Shaper

The neutron pulses coming from the pulse amplifier are further processed by the pulse shaper. The processing performed by the pulse shaper is designed to prepare the Source Range neutron pulses for final processing by the log pulse integrator.

Pulse shaping is a necessary intermediate step, so that the log pulse integrator can function in conjunction with the pulse shaper module to

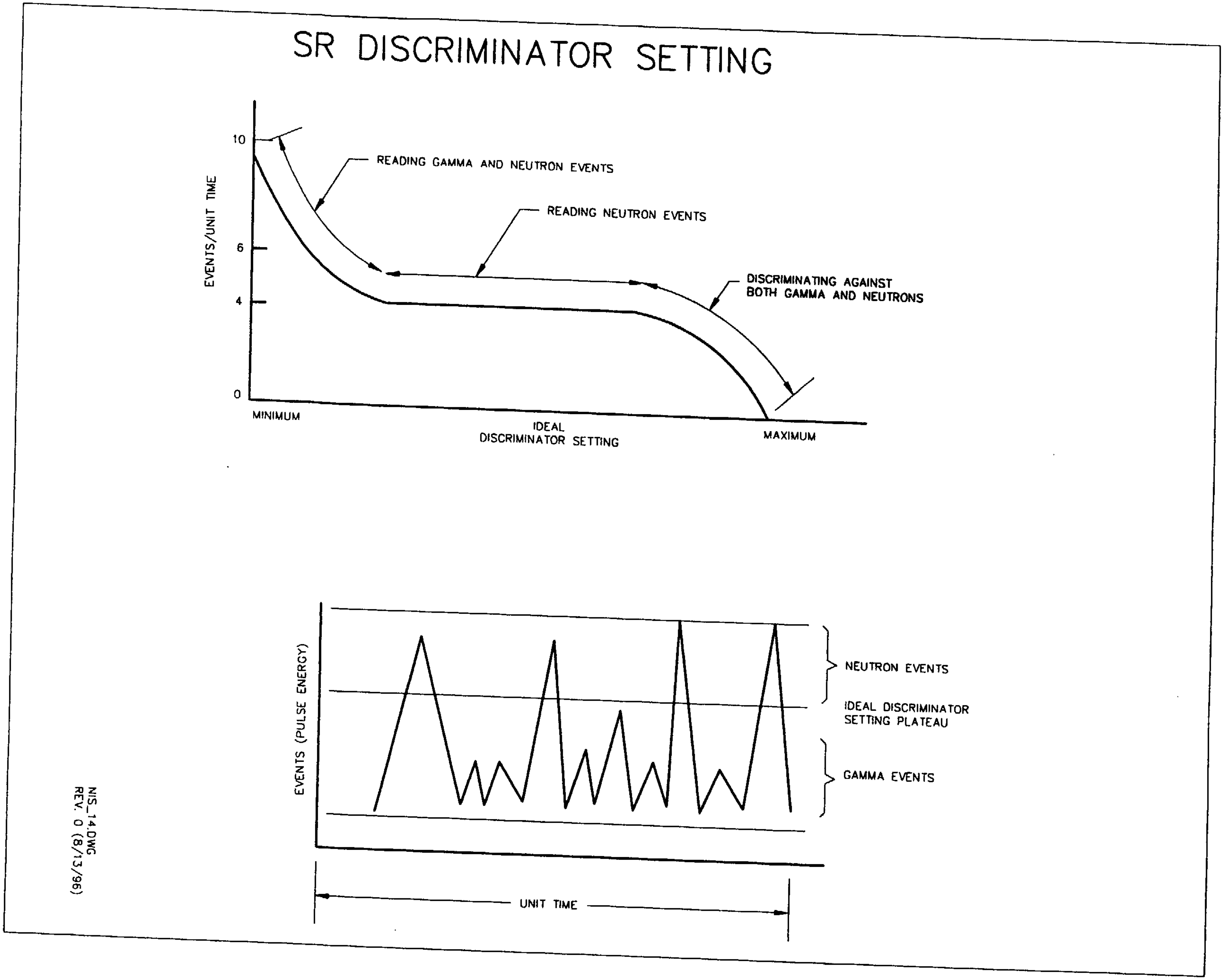


Figure 13-13: SR Discriminator Setting (NIS-14)

W 5 60

WS 60

the inner and outer cylinders), the neutrons react with the boron causing ionization.

$$_{5}B^{10} + _{0}N^{1} \rightarrow (_{5}B^{11})^{*} \rightarrow _{3}Li^{7+++} + _{2}\alpha^{4++} + \text{energy}$$

The lithium and alpha particle resulting from this reaction cause secondary ionization in the outer can. The electrons produced by the ionization are collected on the outer can wall. This produces a signal that is proportional to the neutron flux. Electrons are also collected on the outer can wall from the gamma radiation, which interacts with the outer gas volume. This additional signal is proportional to the gamma flux and is additive to the neutron flux signal. The outer can operates in the Ionization Region; thus, all the charged particles produced in the initial ionizing events are collected on the electrodes.

In the inner can, the gamma flux also reacts with the N_2 gas, producing a signal proportional to the gamma radiation. The inner can is operated in the Recombination Region to permit adjustment of the output current by varying its applied voltage. The inner can voltage is called the compensating voltage. If the compensation voltage is set properly, the outer can signal due to gamma plus neutron flux, will interact with the inner can gamma flux only signal. The gamma signals cancel out leaving the neutron only signal which is then amplified before it is displayed on the meter or sent to the protection and control circuitry.

2.10.2 Gamma Compensation in the Intermediate Range

It is necessary to define the term compensation and the effects of "under-compensation" and "over-compensation" to clearly understand the process of neutron detection in the intermediate range.

Compensation is a term applied to the negative voltage signal applied to the inner can of the CIC which cancels or compensates for the current signal produced by the gamma radiation interacting within the outer can of the detector. This becomes very important to the operator because an incorrect setting of compensating voltage, i.e. over-compensation or under-compensation, would cause an erroneous neutron level indication on the meters, as shown on Figure 13-22.

- Over-compensation occurs when the compensation voltage is set to high. This results in a higher current due to gamma flux in the inner can than is being generated in the outer can due to the same gamma flux. The results of this mismatch is that part of the current due to the neutron flux is also cancelled, causing the indicated current level to be less than actual.

- Under-compensation occurs when the compensation voltage is less than that required. The current due to gamma from the inner can is now smaller than the current due to gamma from the outer can. This allows some current due to gamma in the outer can to remain and add to the neutron flux resulting in increased output current from the detector causing it to indicate above actual levels.

To obtain this true neutron-only signal, the two opposing gamma signals must be cancelled exactly. Since it is physically impossible for both the inner and outer cans to be manufactured identically sensitive to the gamma flux present under all operating conditions, the problem of how to ensure exact compensation arises. By grooving the inner electrode and applying a variable negative voltage, the size of the inner can is adjusted electrically. The inner can of the CIC operates in the recombination region of the detector characteristic curve and, by adjusting the compensating voltage, only a fraction of the total ionization is collected.

The IR drawer monitors reactor power over a range of eight decades between 10^{-11} and 10^{-3} ion chamber amperes. Indications of level and startup rate (SUR) are provided at the NIS cabinets, and on panel FCF.

Because neutron events are occurring at a high rate, no signal conditioning is necessary prior to the log current amplifiers. A block diagram of the intermediate range in provided as Figure 13-23.

2.10.3 Log Current Amplifier

This assembly receives current from the detector in the range between 10^{-11} and 10^{-3} amperes. The assembly provides a logarithmic voltage output, 0 to 10 VDC, proportional to a linear input current. With the use of the log amplifier, the wide range current input is compressed logarithmically to a usable voltage suitable for metering and the generation of trip signals. (Figure 13-23 provides a block diagram of the intermediate range.) The output from the log amplifier is simultaneously coupled to an isolation amplifier and four bistable relay drivers. The output is also displayed on the neutron level meter calibrated in amperes between 10^{-11} and 10^{-3}.amps.

Internal switches and potentiometers are provided for setting and adjusting the log current amplifiers. Both fixed and variable signals can be injected into the log amplifiers for testing and calibration purposes. This is accomplished by the use of switches located on the front panel of the drawer and a calibrate module located inside of the IR drawer assembly.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 61	Group #	2	
	K/A #	028A2.03	
OK	Importance Rating	3.4	

Malfunctions or operations on the HRPS; and based on those predictions use procedures to correct, control or mitigate the consequences of the hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment

Proposed Question: Common 59

Given the following plant conditions:

- From full power, a Large Break LOCA occurred.
- Containment hydrogen concentration is at 2%.
- 31, 32 and 34 Fan Cooler Units started automatically on the Safety Injection
- 31 and 32 Containment Spray Pumps automatically started

Which one of the following actions should be taken to address these conditions?

A. Start 33 and 35 Containment Fan Cooler Units to ensure adequate mixing of Containment atmosphere.

B. Operate at least one of the Hydrogen Recombiners, thereby minimizing the potential for a hydrogen burn.

C. Initiate a containment purge to reduce hydrogen below 1%, thereby minimizing the potential for a hydrogen burn.

D. Allow Containment Spray to continue to run for 4 hours, then resample to see if spray flow has reduced Hydrogen concentration to <1%.

Proposed Answer:

B. Operate at least one of the Hydrogen Recombiners, thereby minimizing the potential for a hydrogen burn.

Explanation (Optional):

Technical Reference(s): ES-1.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-VCPAH2 5 (As available)

Question Source: Bank # INPO 26326
Modified Bank # (Note changes or attach parent)
New

Question History: 8/1/2003 Palisades 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

This procedure should be continued while obtaining hydrogen sample in Step 35.a.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
35.	<u>CHECK Containment Hydrogen Concentration:</u>	
	a. OBTAIN containment Hydrogen concentration: 1) DIRECT Chemist to sample containment Hydrogen 2) PERFORM SOP-SS-4, CONTAINMENT HYDROGEN CONCENTRATION MEASUREMENT SYSTEM	
	b. CHECK containment Hydrogen concentration measurement OBTAINED	b. PERFORM the following: 1) <u>WHEN</u> containment Hydrogen concentration has been obtained, <u>THEN</u> PERFORM Step 35.c and Step 35.d. 2) GO To Step 36.
	c. CHECK Hydrogen concentration - LESS THAN 4.0% BY VOLUME	c. PERFORM the following: 1) CONSULT TSC for additional recovery actions. 2) GO To Step 36.
	d. CHECK Hydrogen concentration - LESS THAN 0.5% BY VOLUME	d. PLACE Hydrogen Recombiner system in service per SOP-CB-7.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 62	Group #	2	
	K/A #	033K4.03	
OK	Importance Rating	2.6	

Knowledge of design features and / or interlocks which provide for anti-siphon devices

Proposed Question: Common 60

Which one of the following statements describes the design feature that prevents inadvertent draining of the spent fuel pit through the spent fuel pit cooling (SFPC) System?

A. SFPC pumps will automatically trip when the low SFP level alarm is annunciated.

B. Deepest SFPC suction piping extends only halfway down into the SFP.

C. SFPC discharge piping has a siphon breaker slightly below the normal water level.

D. Primary makeup valve to the SFP automatically opens on a low level in the SFP.

Proposed Answer:

C. SFPC suction piping has a siphon breaker slightly below the normal water level.

Explanation (Optional):

Technical Reference(s): SD-17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-FHD001 F (As available)

Question Source: Bank # INPO 25137

Modified Bank # (Note changes or attach parent)

New

Question History: 12/4/2002 Millstone 2

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

loaded in the storage racks. The storage racks can be used for storage of new or partially spent fuel.

Per Technical Specification 3.8, water in the pool must be borated to at least 1000 ppm whenever fuel is being moved in the Spent Fuel Pit. This ensures that subcriticality will be maintained during a fuel handling accident when fuel may be dropped onto or next to a rack cell.

It is impossible to insert an assembly into the storage racks in other than a prescribed location. The fuel racks are designed with sufficient center-to-center distances between assemblies to ensure Keff is less than 0.95 if the assemblies are inserted in accordance with Technical Specification 3.8 and no soluble boron is present.

To further ensure that subcriticality is maintained during a refueling, the concentration of borated water used to fill the Spent Fuel Pit matches the Reactor Cavity and Refueling Canal. Daily checks of refueling water boron concentration verify the proper shutdown margin during refueling operations.

The borated Spent Fuel Pit is filled to elevation 93' 8", which is in excess of 23 feet above the fuel and serves as radiation shielding. There are no gravity drains on the Spent Fuel Pit, eliminating the chance of inadvertently draining the Pit. A decrease in Spent Fuel Pit level to 22" below the top actuates the SPENT FUEL PIT LEVEL alarm on Control room panel SKF. Loss of the shielding water from the Spent Fuel Pit is prevented by:

- The suction of the Spent Fuel Pit Cooling Pump is taken six feet below the surface of the Pit.
- The Spent Fuel Pit Cooling System discharges into the pool seven feet above the top of the spent fuel assemblies. This discharge line has a hole drilled in it to prevent it from becoming a siphon and partially draining the pit.
- The cleanup system skimmer pump takes suction from, and discharges to, the surface of the Pit.
- There are no drains on the bottom or side walls of the Spent Fuel Pit.
- The Fuel and Core Component Handling System is designed to seismic Class I criteria.
- The transfer tube is blind flanged and sealed with two O-rings (with weld channel air applied between them) on the Containment side. The gate valve on the spent fuel side also

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 63	Group #	2	
	K/A #	045A1.06	
OK	Importance Rating	3.3	

Ability to predict and / or monitor changes in parameters associated with operating the MT/G system controls including expected response of secondary plant parameters following T/G trip

Proposed Question: Common 61

The following plant conditions exist:

- The plant is operating at 100%.
- All systems are lined up in their normal lineups.
- All control systems are in automatic.
- Main Generator output breakers 1 and 3 trips due to a pilot wire trip.

Which of the following describes the expected immediate plant response?

A. S/G pressure initially increases as main turbine is lost, S/G levels initially decrease due to shrink, feed flow initially increases.

B. S/G pressure initially increases as main turbine is lost, S/G levels initially decrease due to shrink, feed flow initially decreases.

C. S/G pressure initially decreases as main turbine is lost, S/G levels initially decrease due to shrink, feed flow initially increases.

D. S/G pressure initially decreases as main turbine is lost, S/G levels initially increase due to lower steam pressure, feed flow initially decreases.

Proposed Answer:

B. S/G pressure initially increases as main turbine is lost, S/G levels initially decrease due to shrink, feed flow initially decreases.

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICSGL 2 (As available)

Question Source: Bank # INPO 24696
Modified Bank # (Note changes or attach parent)
New

Question History: 5/30/2003 Seabrook 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 64	Group #	2	
	K/A #	055K3.01	
OK	Importance Rating	2.5	

Knowledge of the effects that a loss or malfunction of the CARS will have on the main condenser

Proposed Question: Common 62

Given the following:

- The unit is at 100% power.
- All major controls are in AUTO.
- T_{AVE} has slowly INCREASED 0.2°F in the last 5 minutes.
- Main Generator output has DECREASED 10 MWe.

Which ONE (1) of the following describes the cause of the above indications?

A. SG Safety Valve leakage.

B. Inadvertent RCS dilution.

C. Inadvertent Control Rod insertion.

D. Condenser Air Ejector malfunction.

Proposed Answer:
D. Condenser Air Ejector malfunction.

Explanation (Optional):

Technical Reference(s): SD-20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-CND001 0002 (As available)

Question Source: Bank # INPO 23359
Modified Bank # (Note changes or attach parent)
New

Question History: 3/10/2003 Indian Point 3

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

of LCV 1128 and 1128A. At the zero reference level 1128A is nearly full open and 1128 is starting to OPEN. LCV 1129 is at approximately 50%. Condensate is recirculating through the CST. The controllers are adjusted to maintain constant level in the Hotwell with steady state conditions.

The operator starts condensate pumps as necessary to maintain boiler feed pump suction pressure above 350 psig.

3.5 Abnormal Operation Overview

3.5.1 Loss of condenser vacuum

A loss of condenser vacuum could result from any of the following conditions:

- Loss of condenser integrity (air in-leakage).
- Loss of air ejection capability either through loss of steam supply, operator error, or failure (back firing) of on-line air ejectors.
- Loss of circulating water cooling flow through the condenser tubes.
- Failure of the gland steam regulator or condensate return valve LCV-1133 off the gland sealing steam regulator level control chamber.

Indications to the operator of a loss of condenser vacuum casualty could include any of the following:

- Decreasing condenser vacuum
- Low condenser vacuum alarm
- High turbine exhaust temperatures
- Low gland sealing steam condenser condensate flow rate,
- 6.9 kV motor trip alarm (common),
- Low gland sealing steam pressure,
- Generated plant megawatt decrease.

A loss of condenser vacuum could result in an automatic unit trip (18 inches Hg vacuum). Operation of the hoggers can be very effective in slowing or stopping a decreasing condenser vacuum. The hoggers have a high capacity but are not as efficient.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 65	Group #	2	
	K/A #	072A3.01	
OK	Importance Rating	2.9	

Ability to monitor automatic operation of the ARM system including changes in ventilation alignment

Proposed Question: Common 63

Radiation levels in the Fuel Storage Building (FSB) INCREASED causing R-5, Fuel Storage Building Monitor to reach the alarm setpoint.

By design, which of the following would AUTOMATICALLY occur in the FSB due to this condition?

A. Start air tempering units, shut the sliding door, stop exhaust fans and apply air to door seals.

B. Stop air tempering units, shut the sliding door, place charcoal filters in service and remove air to door seals.

C. Stop air tempering units, shut the sliding door, charcoal filter face dampers open and start exhaust fan.

D. Start exhaust fans, apply air to door seal, place charcoal filter in service and apply air to door seals.

Proposed Answer:

C. Stop air tempering units, shut the sliding door, charcoal filter face dampers open and start exhaust fan.

Explanation (Optional):

Technical Reference(s): SD-12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RMSARM B (As available)

Question Source: Bank # INPO

Modified Bank # 21458 (Note changes or attach parent)

New

Question History: 7/17/2002 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

FSB AREA

INPUT DEVICE: 3RM005

SETPOINT: 100 mR/hr

1.0 CAUSES

1.1 High activity on FSB area monitor

2.0 AUTOMATIC ACTIONS

2.1 FSB supply fans stop and outlet dampers close.

2.2 Exhaust fan starts.

2.3 Inlet and outlet dampers to charcoal filter open.

2.4 Door seals inflate.

2.5 Rolling door closes.

3.0 SUBSEQUENT ACTIONS

3.1 GO TO ONOP-RM-2, High Activity - Radiation Monitoring System.

4.0 REFERENCES

4.1 ONOP-RM-2, High Activity - Radiation Monitoring System changing conditions within the core, such as fuel element failure. The monitor's range is 10^{-1} to 10^{4} mR/hr. R-4 readouts and alarms are local and on a RM-1000 on Panel D-1 in the Control Room.

Exceeding 10 mR/hr actuates "AREA MON HIGH RAD" on Panel SBF-2. An alarm block panel on the wall north of the Flight Panel allows manual blocking of the alarm annunciator so other monitors using the same category alarm can function. Manual pushbuttons block and unblock the annunciator and are backlit when the monitor is blocked.

4.4.4 Channel R-5, Fuel Storage Bldg. Monitor

ARM channel R-5 is at the 95-foot elevation of the Fuel Storage Bldg. It is a double detector with a GM tube for the low range and an ion chamber for the high range. R-5 warns of radiological hazards and of operational problems in the Spent Fuel Pit, such as excessively high activity levels in the pit. The monitor's range is 10^{-1} to 10^{7} mR/hr.

R-5 readouts and alarms are at the detector, on a RM-80 in the Microprocessor Room on the 55-foot elevation of the PAB, on an RM-23A in the RMCC, and on grid five of the Bantam 11.

The alert setpoint is 50 mR/hr and the alarm setpoint is 100 mR/hr. The alarm condition switches the Fuel Building Ventilation System to the Emergency Mode. This consists of shutting down the air tempering units, the exhaust fans starting, the charcoal filters being placed in service, the sliding door shutting and air being applied to the door seals. Exceeding the alarm setpoint actuates "FSB AREA" on the Radiation Monitor Control Cabinet.

4.4.5 Channel R-6, Sampling Room Monitor

ARM channel R-6 is at the 55-foot elevation of the PAB in the Chemical Sampling Room on the north wall. It warns of excessive or increasing radiation fields during sampling operations. The monitor range is 10^{-1} to 10^{4} mR/hr. R-6 readouts and alarms are local and on a RM-1000 on Panel D-1 in the Control Room.

Exceeding 20 mR/hr actuates "AREA MON HIGH RAD" on Panel SBF-2. An alarm block panel on the wall north of the Flight Panel allows manual blocking of the alarm so other monitors using the same category alarm can function. Manual pushbuttons block and unblock the annunciator and are backlit when the monitor is blocked.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 66	Group #	1	
	K/A #	G2.1.1	
OK	Importance Rating	3.7	

Knowledge of conduct of operations requirements

Proposed Question: Common 64

Given the following conditions:

- A licensed reactor operator who has an inactive license has been performing administrative duties in the Training Section for twelve months.
- He is returning to Operations and is to be placed back on shift.

Which ONE (1) of the following are the minimum requirements for returning his license to an active status?

A. Complete 40 (any combination of hours) hours watch standing duties in the RO position, including one shift turnover, and conduct a complete plant tour with the Shift Manager.

B. Complete 56 (7 eight hours shifts) hours watch standing duties in the RO position, including shift turnovers before and after each shift, and conduct a complete plant tour with the Shift Manager.

C. Complete 60 (5 Twelve hour shifts) hours watch standing duties in the RO position, including either the on-coming or off-going shift relief, and review all the procedure changes for the past 7 days.

D. Complete FIVE full normal shifts (8 or 12 hours), including either the on-coming or off-going shift relief, and review all the procedure changes for the past 7 days.

Proposed Answer:

A. Complete 40 (any combination of hours) hours watch standing duties in the RO position, including one shift turnover, and conduct a complete plant tour with the

Shift Manager.

Explanation (Optional):

Technical Reference(s): OAP-032 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: IOWKB-ILO-ADM-001 0001 (As available)

Question Source: Bank # INPO 28112
Modified Bank # 28112 (Note changes or attach parent)
New

Question History: 9/27/2004 Robinson 2

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

WS 66

4.1.5 License Reactivation:

4.1.5.1 To restore an Inactive/Current License to active status perform the following and document using Attachment 1, INACTIVE LICENSE REACTIVATION FORM:

a) Complete a minimum of 40 hours of on-shift time under the direction of an active RO or SRO licensee, in the position, which the individual will be assigned.

1) Conduct a complete tour of the plant with the SM.

2) Conduct a Shift Turnover.

3) The licensee may manipulate controls under the direction of an active licensed operator.

b) The 40 hours of on-shift watch time must be completed within a single calendar quarter, and may be any combination of hours, which add up to 40 (NOT required to be consecutive).

c) In addition to completing Attachment 1, INACTIVE LICENSE REACTIVATION FORM documentation of on-shift time towards license reactivation is included.

d) Obtain certification from the Supervisor-Licensed Operator Requal the licensee is satisfactorily up-to-date on License Operator Requalification Program training requirements.

e) ENSURE Respiratory Protection and qualification are met.

f) Upon satisfactory completion of the above, the AOM may certify that the individual meets the requirements to restore the license to active status.

g) Forward completed Inactive License Reactivation Forms to the Superintendent of Operations Training for proper retention.

4.1.5.2 To restore a Inactive/Current SRO License to active status for fuel handling duties (RSRO), the Inactive/Current SRO SHALL stand one 8-hour watch supervising refueling activities in the VC under the direction of a current SRO, and complete Attachment 1, INACTIVE LICENSE REACTIVATION FORM, in accordance with step 4.1.5.1c) through 4.1.5.1g) **{Reference 7.1.2}.**

4.1.5.3 To restore a license removed from licensed duties to active status perform the following and document using Attachment 2, REMOVED FROM LICENSED DUTIES REACTIVATION FORM

a) Obtain certification from the Medical Review Officer of the satisfactory completion of medical requirements, if applicable.

b) Obtain certification from the Supervisor Licensed Operator Requal of completion of Licensed Operator Requalification Program requirements, if applicable.

c) Complete watch standing proficiency requirements for the current calendar quarter in accordance with section 4.1.2.

d) Upon satisfactory completion of the above, the AOM may certify that the individual meets the requirements to restore the license to active status.

e) Forward completed Removed from Licensed Duties Reactivation Forms to the Superintendent of Operations Training for proper retention.

4.1.6 *Changes in NRC License Status*

4.1.6.1 The AOM – Training SHALL initiate notification of the NRC of a change in license status resulting from the following:

a) A licensee is determined to no longer have a continued need for the license. [10CFR 50.74(a)].

b) A licensee is convicted of a felony. [10CFR 50.74(d)].

4.1.6.2 To ensure NRC notification is performed within 30 days, the AOM - Training SHALL initiate an LOCR item with a scheduled completion date of 14 days from the date of determination to the Licensing Manager.

4.1.6.3 The Human Resources Manager initiates notification to Licensing Department of change in license status as a result of termination of employment.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 67	Group #	1	
	K/A #	G2.1.32	
OK	Importance Rating	3.4	

Ability to explain and apply all system limits and precautions

Proposed Question: Common 65

What is the BASIS for the requirement to ensure that the temperature in the steam generators is less than or equal to the temperature in the Reactor Coolant System (RCS) cold leg PRIOR to directing initial starts of Reactor Coolant Pumps?

A. Ensures an available heat sink for the RCS when securing shutdown cooling.

B. Prevents a rapid depressurization of the steam generators due to a cooldown.

C. To limit the thermal stresses experienced by the steam generator tubes.

D. Ensures that heat energy addition to the RCS from the steam generators does not occur, causing rapid pressure rise in RCS.

Proposed Answer:

D. Ensures that heat energy addition to the RCS from the steam generators does not occur, causing rapid pressure rise in RCS.

Explanation (Optional):

Technical Reference(s): 3-SOP-RCS-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RCSRCP H (As available)

Question Source: Bank # INPO 20893
Modified Bank # (Note changes or attach parent)
New

Question History: 12/21/2001 Palisades 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 2

Comments:

W567

NOTE

- RCP 31, 33 and 34 bearing lift oil minimum pressure of 500 psig is verified by applicable RCP Bearing Lift Press white permissive light illumination on Panel SAF.
- RCP 32 minimum bearing lift oil pressure of 500 psig is checked locally (Ref 5.2.12).

____ 4.1.9.3 VERIFY minimum bearing lift oil discharge pressure of 500 psig.

CAUTION

WHEN torque of 750 ft-lbs is exceeded, THEN RCP SHALL NOT be started.

____ 4.1.9.4 REQUEST Maintenance to bar over selected RCP(s).

____ 4.1.9.5 WHEN RCP has been barred over, THEN STOP bearing lift pump. (Panel SAF)

____ 4.1.9.6 REPEAT Step 4.1.9 as necessary for additional RCPs.

____ 4.1.10 IF RCS cold leg temp (TCOLD) is equal to or less than 332°F, THEN VERIFY at least 1 of the sub-steps in Attachment 3, RCP Start Requirement Determination, is satisfied to meet requirements of SR 3.4.12.8 and SR 3.4.12.9.

CAUTION

Do NOT bump or start RCPs with any SG temp greater than any RCS loop temp (THOT or TCOLD), unless Reactor Engineering has evaluated potential RCS heatup effects **{Reference 5.1.3}**.

____ 4.1.11 IF any SG secondary side temp needs to be lowered to less than RCS temp, THEN DRAIN and FILL applicable SG(s) as necessary using either of the following:

____ • SOP-SG-002B, Steam Generator Draining And Dry Layup

____ • SOP-SG-002C, Steam Generator Filling, Chemical Addition, And Wet Layup

____ 4.1.12 ENSURE motor starting times of SOP-EL-004A are met by reviewing Unit Log.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 68	Group #	2	
	K/A #	G2.2.1	
OK	Importance Rating	3.7	

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could effect reactivity

Proposed Question: Common 66

The following plant conditions exist:

- A reactor startup is in progress
- Estimated Critical Position (ECP) is 80 steps on Control Bank D
- The ATC has just commenced CBA rod withdrawal
- Initial count rate prior to shutdown bank withdrawal was 4E1 cps on source range N31 and N32
- Current count rate is stable at 3E2 cps on source range N31 and N32

Which ONE of the following is the next action to be taken?

A. Continue reactor startup and continue to plot source range counts to criticality.

B. Insert ALL control rods and evaluate the ECP.

C. Stop control rod withdrawal until abnormality is understood and does not jeopardize plant safety.

D. Begin emergency boration to achieve 1% shutdown margin.

Proposed Answer:

B. Insert ALL control rods and evaluate the ECP.

Explanation (Optional):

Technical Reference(s): 3-POP-1.2, Attachment 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-POP-006 E-3 (As available)

Question Source: Bank # INPO
Modified Bank # 27237 (Note changes or attach parent)
New

Question History: 5/10/2004 Davis-Besse 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5

Comments:

1.0 PURPOSE

1.1 This procedure establishes the requirements for performing the following:

- Maintaining Mode 3 conditions.
- Entering Mode 2 to:
  - Achieve criticality by withdrawal of control rods.
  - Establish conditions for achieving criticality by RCS Boron dilution.

1.2 This procedure applies to start-up of the reactor from the Mode 3 condition to the Mode 2 condition at approximately 10^{-6} Amps power level.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Reactivity changes SHALL be closely monitored by observation of different parameters such as NIS, Tavg, Tref, Control Rods, and ΔT. **{Reference 5.1.4}**

2.2 Positive Reactivity additions require CRS or SM approval and SHALL be made incrementally. **{Reference 5.1.4}**

2.3 During the approach to criticality, Boron dilution and rod withdrawal SHALL <u>NOT</u> be performed simultaneously.

<u>NOTE</u>

Boron changes may require as long as 30 minutes to affect Source Range count rate.

2.4 Criticality SHALL be anticipated during the following:

- Rod withdrawal
- Boron dilution

2.5 <u>IF</u> count rates indicate a premature approach to criticality during control rod withdrawal, <u>THEN</u> STOP control rod withdrawal. <u>WHEN</u> the source of the abnormality is understood <u>AND</u> does <u>NOT</u> jeopardize plant safety, <u>THEN</u> CONTINUE control rod withdrawal.

2.6 <u>IF</u> either source range channel count rate increases by a factor of two or more during any step involving Boron concentration change, <u>THEN</u> the operation SHALL be immediately suspended until the discrepancies have been resolved.

2.7 An unanticipated second doubling during Shutdown Bank withdrawal SHALL be evaluated by the Shift Reactor Engineer. (Refer to Attachment 2, Action Guidelines)

2.8 IF unexpected situations arise NOT covered in Attachment 2, Operator Action Guidelines, THEN PLACE the Reactor in a safe condition.

2.9 Based on the length of shutdown time, the Source Range NI count rate may initially be dominated by secondary source neutrons, causing reactor criticality to be predicted non-conservatively.

2.10 Steady state startup rate SHOULD NOT exceed 0.5 decade per minute (DPM) and SHALL NOT exceed 1.0 DPM.

2.11 The Reactor SHALL be manually tripped any time MTC becomes positive with the Reactor critical, except during zero power physics testing per ITS 3.1.8.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 69	Group #	2	
	K/A #	G2.2.11	
	Importance Rating	2.5	

Knowledge of the process for controlling temporary changes

Proposed Question: Common 67

Which of the following temporary installations in the plant would require a temporary modification tag to be installed per EN-DC-136, Temporary Modifications?

A. A temporary heater being plugged into a 120V outlet by the RWST level instrumentation to maintain fluid temperature.

B. A leaking relief valve being gagged shut on the Auxiliary Boiler steam line.

C. A hose being connected from a Service Air connection to a temporary sump pump on the 5ft. el of the turbine building.

D. Plastic sheeting being installed over a MOV to protect it from a leaking roof.

Proposed Answer:

B. A leaking relief valve being gagged shut on the Auxiliary Boiler steam line.

Explanation (Optional):

Technical Reference(s): EN-DC-136, Attachment 9.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

Question Source: Bank # INPO 27113
Modified Bank # ____________________ (Note changes or attach parent)
New ____________________

Question History: 7/16/2004 Millstone 3

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____________________

10 CFR Part 55 Content: 55.41 10
55.43 3

Comments:

Part I - SCREENING

Individuals who identify activities that could be a potential Temporary Modification should answer the following questions:

- Is the SSC affected by this activity (or associated SSC) in service, in standby or required to be operable (i.e., not part of a tag-out or considered a maintenance activity)?
- Does the intended activity temporarily modify the SSC configuration so that it deviates from the approved drawings or other design documents or changes to the design function of the SSC?
- Would a design change or equivalency evaluation be required to make the altered configuration permanent?

IF the answer to ANY of the above questions is "NO",
THEN this activity is not a Temporary Modification and this procedure does not apply.

IF the answers to ALL the above questions are "YES",
THEN the activity is a potential Temporary Modification. The individual then should use Part II - Exclusions, below, to determine if the activity should be implemented using the Temporary Modification process.

Industry documents [e.g., INPO TS-412] define Temporary Modifications as temporary activities that alter a system's configuration. This guideline provides a global understanding of where the boundaries are and how to apply the criteria consistently and clearly include a review of other SSCs that are relying on the affected SSC to remain operable. Based on experience, it is evident that real life activities create conditions that can sometimes be equally justified on either side of the Temporary Modification definition. Determinations must consider work control and safety aspects and should default to the conservative.

Part II – EXCLUSIONS

NOTE
Temporary activities, including those that are excluded below, should be short term in nature. Aggressive actions shall be taken to remove any long term temporary configuration.

The following are temporary activities that are excluded from the requirements of this procedure:

1. Inoperable, abandoned or out of service equipment controlled by an active Work Order or clearance, provided the alteration does not affect adjacent SSCs, and is removed prior to declaring the SSC operable.

2. Degraded or nonconforming SSC that is being tracked by a Condition Report. This interpretation is consistent with the 10CFR50.59 program requirements. Inherent to the definition of a nonconforming condition, the SSC does not conform to its design conditions. The continued operation is justified by an operability assessment and the condition is expected to be corrected in a timely manner, commensurate with the System Sturcture or Component's safety function. However, specific temporary physical plant alterations required for compensatory actions to maintain operability would be a Temporary Modification.

Part II – EXCLUSIONS (Continued)

3. Installations during the implementation of an approved permanent plant change. These are controlled by the installation instructions provided:

   a. The installation phase configuration was evaluated by the design document, **AND**

   b. The Process Applicability Determination addresses the installation of the change, **AND**

   c. Second verification of installation and removal are included, if appropriate, in the installation instructions.

4. Troubleshooting Work Orders that temporarily alter a SSC and, provided:

   a. The Work Order contains installation instructions, restoration instructions, verifications, and tracking, **AND**

   b. The Work Order impact summary identifies no adverse effect on associated SSC operational requirements, **AND**

   c. The SSC is restored, or a temporary modification installed, prior to declaring the SSC operable.

5. Isolation of equipment using system or component design features; e.g., isolation valves, breakers, fuses, spectacle flanges, hand-jacks on AOVs, installed bypass/defeat mechanisms for protective Engineered Safety Features channels.

6. Replenishing consumables on operating equipment when SSC design permits these activities without impairing function; e.g., changing filters, adding oil, grease, coolant, electrolyte, freon, etc. [It is assumed that the activity is of limited duration and the SSC is attended while the consumable is being replenished.]

7. Equipment checks on operating equipment when equipment is designed to permit these checks without impairing function; e.g. checking electrolytic level, specific gravity & temperature; oil level, oil or other sample acquisition, fuse status, etc. using doors, inspection ports, battery cell caps, dipsticks, etc. [It is assumed that the activity is of limited duration and the SSC is attended while the check is being performed.]

8. Temporary structures that do not impact any structural design requirements and are in accordance with the storage guidelines; including seismic interaction, HELB, fire protection, fire loading, masonry walls, flooding (internal and external), floor loading, pressure boundaries, operational access. The impact review of a temporary structure should be documented under the ER Process.

Part II – EXCLUSIONS (Continued)

9. Removal of seismic class I supports or other components when required to permit maintenance functions, provided the removal has been evaluated by Engineering, documented in an Engineering Request and found acceptable and any restrictions stated in the evaluation are complied with; e.g., allowable analyzed outage time.

   Note: Since the removal of the seismic class I support may not necessarily be directly related to the SSC for which the maintenance activity is being performed, the engineering evaluation should consider the impact to the SSC affected by the support removal only.

10. Activities affecting interlocks, indications, annunciators, etc. to in-service or standby SSCs which do not have technical specification requirements, when required to permit maintenance functions [test, surveillance, reactor physics testing, calibration, hydrostatic test, repair, refurbishment, replacement with equivalent component, etc.] that restores the SSC to its original configuration via the work control process. Activities should not be credited with maintaining associated SSCs in service or technical specification "operable". The time limitations should be in accordance with work control procedures.

    Example:
    Calibration of a condensate pump discharge pressure instrument loop results in temporary loss of the high discharge pressure trip interlock.

11. The lifting of a wire or removal of a circuit board to perform troubleshooting or an electrical test in accordance with approved procedures or work instructions, provided the wire or circuit board is reconnected immediately following the test, lifting/removal is documented, and restoration is documented and verified through independent or concurrent verification.

12. Hand-Held Jumpers physically held in place by an individual, controlled by approved procedures or work instructions within the Work Orders Process.

13. Use of electrical power receptacles and fusible disconnects for tools, maintenance equipment, and other devices not directly involved with plant operation. Temporary devices required to maintain SSCs in service, in standby or operable are <u>not</u> covered by this exclusion.

14. Connection of gauges to Non-Quality related vents, drains or pressure points when installation and removal is tracked and controlled by approved procedures or work instructions, and the gauge is isolable and remains isolated except when readings are taken.

15. Use of temporary test gauges in performance of repetitive surveillance or maintenance activities on operable equipment when the installation is attended until the gauge is isolated. The installation and removal of the test gauges must be tracked by the controlling process [i.e., procedure, etc.]

16. Test equipment installed in a totally non-intrusive manner [including seismic and structural considerations].

17. Use of permanently installed vent or drain connections including the addition of required fittings. This <u>does not</u> include cross-tying vents/drains from two locations or using the vent/drain for another function.

ATTACHMENT 9.2 **TEMPORARY MODIFICATION SCREENING AND EXCLUSIONS**

Sheet 4 of 4

Part II – EXCLUSIONS (Continued)

18. Process connections used in accordance with its designed intended function; i.e., use of sampling equipment, service air, breathing air, domestic water, or deionized water.

10. Connection of chemistry sampling equipment to Non-Quality related vents or drains in a manner that does not allow unattended pressurization of the connection.

20. Hoses, rubber patches, tubing and catch-containment devices used to reduce the spread of contamination, or for housekeeping purposes, regardless of component or system ISI and safety class, only if the device:

    a. Is **NOT** attached to pressure relief devices; **AND**
    b. Is **NOT** required to maintain or restore component or system functionality or operability; **AND**
    c. Does **NOT** alter or degrade component or system structural integrity; **AND**
    d. Is **NOT** credited with restoring pressure boundary; **AND**
    e. Is easily reversible.

21. Removal of protective relays for calibration. Removal is acceptable only if:

    a. The relay is designed for in-service removal,
    b. Adequate protection is provided by remaining relays, and calibration activity is of limited duration.

22. Pulling of an annunciator card for an alarm which is associated with inoperable, abandoned and/or physically removed, or out of service equipment, unless the removed annunciator card will mask an alarm associated with in-service or operable equipment, as long as there is an open Work Order/Work Request against the associated SSC.

23. Removal of Continuous Vibration Monitoring Equipment Channels and Circuits from service in accordance with site-specific procedures.

24. Bearing monitor and computer alarms taken out of scan in accordance with site procedures.

25. Temporary set-point changes in accordance with site-specific procedures.

26. Leak repairs, including enclosures around a pipe or component that will not impact component function (e.g., ENN-ME-S-001 or site procedures for additional guidance).

27. Bypassing a failed alarm for in-service and/or operable equipment, where alternate currently installed means of monitoring the same parameter, as the failed alarm input, are available (e.g.: one of the five waterbox conductivity elements are in alarm and is bypassed with the remaining elements providing redundant indication.)

28. Activities to support maintenance as defined in NEI 96-07 revision 1. (See section 4.1.2 in NEI 96-07, which is available on ENN Engineering webpage under 50.59 Information or on the ENS 50.59 webpage.)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 43	Group #	1	
	K/A #	039K1.07	
OK	Importance Rating	3.4	

Knowledge of the physical connections and / or cause-effect relationships between the MRSS and the AFW system

Proposed Question: Common 68

Unit 3 was operating at 100% power when a steam line break occurred in the Aux Feed Building. The break is upstream of MS-42, 33 Steam Generator Supply to 32 ABFP.

During the performance of E-2, Faulted Steam Generator Isolation, the NPO fails to manually shut MS-42 as directed by the control room.

Which of the following describes a consequence of this error?

A. 32 ABFP will lose its steam supply because both Steam Generators will blow down through the rupture.

B. 32 ABFP will lose its steam supply because PCV-1139, 32 ABFP pressure control valve, will automatically close.

C. 32 ABFP will NOT be affected because MS-42, 33 Steam Generator Supply to 32 ABFP, is a stop check valve.

D. 32 ABFP will NOT be affected because MS-1-33, 33 Steam Generator MSIV will be closed by the control room.

Proposed Answer:

C. 32 ABFP will NOT be affected because MS-42, 33 Steam Generator Supply to 32 ABFP, is a stop check valve.

Explanation (Optional):

Technical Reference(s): SD-21.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-AFW001 0002 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: 9/29/2003 Point Beach 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2,9
55.43

Comments:

pressure equal to the minimum pressure setting of the main steam safety valves. Design shutoff heat is 3750 feet.

The minimum NPSH required for TD-ABFP operation is 16 feet. Actual available minimum value has been calculated to be 66 feet.

The #32 ABFP is driven by an axial flow non-condensing turbine (Figure 21.2-6) rated at 970 HP and supplied by main steam from SGs #32 and #33. The Low Pressures Exhaust pipe drain, MS-114, is normally cracked open to provide a path for condensate to drain from the turbine casing. The pump and turbine have bearing coolers fed off the pump discharge.

This pump can feed all SGs. It is normally controlled manually from the panel SCF in the control room.

The turbine exhaust of this non-condensing turbine is directed to atmosphere. The turbine is provided with a missile shield to protect the suction piping and other important equipment in the ABFP room from flying blades.

2.5.1 Turbine Driven Auxiliary Boiler Feed Pump Steam Supply

The MSS provides steam to the Auxiliary Feedwater Pump Turbine (ABFPT) through a steam line fed by Steam Generators 32 and 33 as shown in Figure 21.2-3. Each four inch steam line has a stop check valve, MS-41 from SG 32 and MS-42 from SG 33. These stop check valves are located in the "steam bridge" area of the ABFP building. The stop check valves perform the following functions:

- prevents a fault on either steam generator from preventing operation of the feed pump
- prevents the steam supply to the 32 ABFP from becoming a source of steam to a faulted steam line.

Each four inch steam line is supplied by a short section of three inch piping tapping off the main steam header. These three inch sections function as a flow restrictor. This limits the combined steam flow from 33 and 32 steam lines to less than that of a stuck open main steam safety valve. These valves are locked open when the AFWS is required to be operable.

Downstream of the stop check valves, the steam lines combine. On this common line are two, air to close, spring to open, isolation valves, PCV 1310A & 1310B. These valves isolate the steam supply to the 32 ABFP in case of a steam line break in the ABFP room. With the valve switch in AUTO, a high temperature in the room of 130°F will close both valves. This feature is designed to preserve the electrical

		RO	SRO
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 70	Group #	2	
	K/A #	G2.2.28	
OK	Importance Rating	2.6	

Knowledge of new and spent fuel movement procedures

Proposed Question: Common 69

Which ONE (1) of the following, as a minimum, shall be present in the Containment to monitor all activities during Core loading or unloading?

A. VC Coordinator

B. Reactor Engineer

C. Shift Technical Adviser (STA)

D. Refueling Senior Reactor Operator (SRO)

Proposed Answer:
D. Refueling Senior Reactor Operator (SRO)

Explanation (Optional):

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # INPO 28116

Modified Bank # ______________ (Note changes or attach parent)

New ______________

Question History: 9/27/2004 Robinson 2

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______________

10 CFR Part 55 Content: 55.41 ______

55.43 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 71	Group #	3	
	K/A #	G2.3.10	
OK	Importance Rating	2.9	

Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure

Proposed Question: Common 70

Given the following conditions at a work site:

- A task is required to be performed in a Radiation area
- Radiation level is 40 mrem/hr
- Radiation level with shielding is 10 mrem/hr
- Time for one worker to install and remove shielding is 15 minutes
- Time to conduct the task with one worker is 1 hour
- Time to conduct the task with two workers is 20 minutes

Assumptions:

- A dose rate of 40 mrem/hr will be received while installing and removing the shielding.
- Shielding is installed and removed by one worker only.

Which ONE of the following would result in the lowest total whole body dose?

Conduct the task with:

A. two workers without shielding.

B. one worker without shielding.

C. two workers with shielding.

D. one worker with shielding.

Proposed Answer:

C. two workers with shielding.

Explanation (Optional):

Technical Reference(s): ______________________ (Attach if not previously provided)

____

____

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # INPO 27008

Modified Bank # ______________ (Note changes or attach parent)

New ______________

Question History: 12/15/2003 Turkey Point 3

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 72	Group #	3	
	K/A #	G2.3.9	
OK	Importance Rating	2.5	

Knowledge of the process for performing a containment purge

Proposed Question: Common 71

Given the following:

- A stuck open PZR Safety Valve caused a Reactor Trip AND Safety Injection
- The PRT Rupture Disk ruptured
- The reactor is now in Cold Shutdown.
- A containment purge using the Containment Purge System is being initiated.
- Containment pressure is 0.8 psig.

Which ONE of the following requirements and/or limitations applies concerning the Containment Purge during this evolution?

A. Notification of the NRC is required.

B. A gaseous discharge permit is required.

C. At least one train of Auxiliary Building Ventilation System shall be in operation.

D. A Containment Purge Exhaust Fan must be started prior to a Containment Vent Exhaust Fan.

Proposed Answer:
B. A gaseous discharge permit is required.

Explanation (Optional):

Technical Reference(s): 3-SOP-CB-003 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-VCVCB 6 (As available)

Question Source: Bank # INPO 28436
Modified Bank # (Note changes or attach parent)
New

Question History: 11/15/2004 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 9

Comments:

CAUTION

- CB Purge System SHALL NOT be operated in Modes 1, 2, 3, or 4.
- WHEN in Mode 6 AND Containment Closure is set,
  THEN starting and stopping Purge Supply Fan may affect VC pressure:
  - o WHEN changing purge Supply Fan status, THEN exercise caution to preclude transferring Refueling Cavity water from VC to FSB due to atmospheric ΔPs. **(Reference 5.2.2)**

NOTE

Only section(s) of 3-PT-Q075A, Channel Functional Test RM 11/12 for testing auto closure of CB purge isolation valves is required to be performed.

4.2 CB Purge System

____ **4.2.1 Startup**

____ 4.2.1.1 IF required,
THEN PERFORM 3-SOP-WDS-013, Gaseous Waste Release Permits.

____ 4.2.1.2 IF auto closure of VC purge isolation valves has NOT been tested in last 90 days,
THEN PERFORM 3-PT-Q075A, Channel Functional Test RM 11/12 prior to entering Mode 6.

____ 4.2.1.3 ENSURE the following valve Individual Air Supply Valves are unlocked and open:

____ • Air Supply Isolation Valve to FCV-1170 Operator (inside VC under 95' grating at purge valves)

____ • Air Supply Isolation Valve to FCV-1171 Operator (at purge valves outside containment)

____ • Air Supply Isolation Valve to FCV-1172 Operator (inside VC under 95' grating at purge valves)

____ • Air Supply Isolation Valve to FCV-1173 Operator (at purge valves outside containment)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 73	Group #	4	
	K/A #	G2.4.18	
OK	Importance Rating	2.7	

Knowledge of the specific bases for EOPs

Proposed Question: Common 72

The following conditions exist:

- A reactor trip has occurred due to a loss of MFW.
- FR-H.1, Response to a Loss of Secondary Heat Sink is in progress.
- The RCS is in a Bleed-and-Feed condition with RCS Temperature stable at 570°F.
- 31 and 32 SG WR levels are off scale low
- 33 and 34 SG WR levels are 10%
- The operators restore a feedwater source and prepare to feed the S/Gs which are dry.
- The CRS directs the operator to establish feed water to only one S/G.

Which one of the following describes the reason for feeding only one S/G under these conditions?

A. To ensure that if a S/G failure occurs due to excessive stresses, the failure is isolated to one S/G.

B. To prevent a rapid cooldown of the RCS that could lead to a pressurized thermal shock condition.

C. To demonstrate the reliability of the FW source before filling all of the steam generators.

D. To determine if one S/G is capable of maintaining adequate heat sink so that RCS bleed-and-feed can be terminated.

Proposed Answer:

A. To ensure that if a S/G failure occurs due to excessive stresses, the failure is isolated to one S/G.

Explanation (Optional):

Technical Reference(s): FR-H.1, Foldout Page (Attach if not previously provided)
FR-H.1, Bases

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPFRH 7 (As available)

Question Source: Bank # INPO 24409
Modified Bank # (Note changes or attach parent)
New

Question History: 9/17/2002 Summer 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

FOLDOUT PAGE

1. BLEED AND FEED CRITERION:

IF the following conditions exist AND bleed and feed has NOT been initiated, THEN immediately GO To FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Procedure Section, Step 9, to initiate bleed and feed:

- Average of the 3 lowest WR SG levels is less than 25% [30%]
- RCS pressure is greater than highest SG pressure

2. SG FEEDFLOW LIMITATIONS:

IF any WR SG level is less than 12% [16%] AND RCS hot leg temperature is greater than 550°F, THEN PERFORM the following for affected SG(s):

- IF RCS bleed and feed has been initiated AND RCS temperatures are increasing, THEN LIMIT use of affected SGs to only one and feed that SG at maximum rate.
- IF RCS temperatures are stable OR decreasing, THEN LIMIT feedflow to affected SG(s) to less than or equal to 100 gpm and all flow changes must be made slowly.
- IF the affected SG(s) with feedflow has a fault or rupture AND any other SG is available, THEN ISOLATE faulted or ruptured SG.

damage. Therefore, based on the above arguments, feed and bleed is not recommended to provide an alternative heat removal method during a loss of secondary heat sink condition.

2.4 Feeding of a Hot, Dry Steam Generator

During restoration of secondary heat sink, it may become necessary to establish feedwater to a hot, dry steam generator. A hot, dry steam generator is defined as a steam generator in which the primary side of the steam generator tubes is above 550°F* and the secondary side has no liquid inventory. The primary side SG tube temperature is determined from hot leg temperature readings. Reestablishment of feedwater is the more desirable mode of recovery from a loss of secondary heat sink than remaining on bleed and feed and establishing cold leg recirculation for long term cooling. However, care must be taken if feedwater is to be reestablished to a hot, dry steam generator.

Since the heat removal capability of one steam generator is always greater than decay heat, it is advisable to reestablish feedwater to only one steam generator regardless of the size of the plant or number of loops. Thus, if a failure in an SG occurs due to excessive thermal stresses, the failure is isolated to one steam generator.

---

* 550°F is a temperature evaluated to be low enough that thermal stress would not lead to a failure when feedwater is established to any remaining dry steam generator.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 74	Group #	4	
	K/A #	G2.4.1	
OK	Importance Rating	4.3	

Knowledge of EOP entry conditions and immediate action steps

Proposed Question: Common 73

Given the following:

- The plant is at 100% power.
- A total loss of 480V power occurs.

Which ONE of the following describes the correct procedure and immediate operator action?

A. E-0, Rx Trip or Safety Injection, ensure Rx is tripped by all Rod bottom lights lit

B. ECA-0.0, Loss of All AC Power, ensure Rx tripped by neutron flux decreasing and close all MSIVs.

C. ES-0.1, Reactor Trip Response, when SI is NOT required, attempt to restore 480V power.

D. ECA-0.0, Loss of All AC Power, ensure Rx tripped by all Rod bottom lights lit and manually trip the turbine.

Proposed Answer:

B. ECA-0.0, Loss of All AC Power, ensure Rx tripped by neutron flux decreasing.

Explanation (Optional):

Technical Reference(s): AOP-12 (Attach if not previously

ECA-0.0 provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOPROU 8 (As available)
I3LP-ILO-EOPC00 6

Question Source: Bank # INPO 28437
Modified Bank # (Note changes or attach parent)
New

Question History: 11/15/2004 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5

Comments:

WS 74

4.2.5 Several procedures contain steps which are designated as Immediate Action steps.

4.2.5.1 The operator is required to memorize all immediate actions AND SHALL be capable of performing all immediate action steps from memory.

4.2.5.2 The following procedures contain immediate action steps:

- E-0 REACTOR TRIP OR SAFETY INJECTION (Steps 1-4)
- FR-S.1 RESPONSE TO NUCLEAR GENERATION/ATWS (Steps 1-2)
- ECA-0.0 LOSS OF ALL AC POWER (Steps 1-2)

4.2.5.3 While operators are performing immediate actions from memory, the CRS SHALL obtain the procedure and restart the procedure performance with the team at Step 1.

4.2.5.4 Transitions OR unexpected conditions SHALL be called out during performance.

4.2.5.5 As a general rule, operators should refrain from taking any additional actions, prudent or not, until the immediate actions are complete. The only exceptions are those actions specifically directed to be performed by AOPs/ONOPs.

4.2.6 Action steps are written such that an operator would normally proceed directly down the left-hand column only. The column contains all the expected conditions, actions, and checks required to accomplish the stated purpose.

4.2.7 The left-hand column contains a high-level statement which describes the task to be performed. This is column is titled "ACTION/EXPECTED RESPONSE". It is called the AER column.

4.2.8 IF sequence of performance is important, THEN the sub-tasks are designated by letters (or numbers if finer detail is provided). Step sequence SHALL be strictly adhered to.

4.2.8.1 IF sequence of performance is not important, THEN the sub-tasks are designated by bullets.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- CSF Status Trees should be monitored for information only. FRPs should NOT be implemented.
- Normal communication channels may be unavailable without AC power. Radios should be used by watch personnel outside the Control Room.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**VERIFY Reactor Trip:** • Reactor trip and bypass breakers - OPEN • Neutron flux - DECREASING	PERFORM the following: a. Manually TRIP Reactor. b. IF Reactor will NOT trip, THEN PERFORM the following: 1) INSERT control rods in auto or manual while continuing performance of this procedure. 2) DISPATCH NPO to trip reactor in accordance with posted operator aid.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	<u>ISOLATE Main Steam:</u>	
	a. Manually CLOSE all MSIVs	a. PERFORM the following: 1) VERIFY all turbine stop valves are closed. 2) DISPATCH NPO to close MSIVs per SOP-ESP-1.
	b. CHECK MSIV bypass valves - CLOSED	b. DISPATCH NPO to close all MSIV bypass valves, if required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
WS # 75	Group #	4	
	K/A #	G2.4.7	
OK	Importance Rating	3.1	

Knowledge of event based EOP mitigation strategies

Proposed Question: Common 74

Given the following conditions:

- A cooldown and depressurization of the RCS is in progress as directed by ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel.
- A Yellow path is noted for Inventory that directs the crew to FR-I.3, Response to Voids in Reactor Vessel.
- The decision is made to continue with the actions of ES-0.3 and NOT transition to FR-I.3, Response to Voids in Reactor Vessel

Why would a transition to FR-I.3 NOT be made?

A. FR-I.3 would only be entered prior to performing a cooldown and depressurization.

B. FR-I.3 addresses voids resulting from non condensable gas evolution, NOT from steam void formation.

C. Upper head steam voiding is expected in these conditions and accounted for in ES-0.3.

D. The Status Trees are monitored "for information only" in these conditions.

Proposed Answer:

C. Upper head steam voiding is expected in these conditions and accounted for in ES-0.3.

Explanation (Optional):

Technical Reference(s): FR-I.3 (Attach if not previously provided)
ES-0.3

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # INPO 26894
Modified Bank # (Note changes or attach parent)
New

Question History: 2/2/2004 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5

Comments:

WS75

Number:	Title:	Revision Number:
FR-I.3	RESPONSE TO VOIDS IN REACTOR VESSEL	12

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

IF A CONTROLLED NATURAL CIRCULATION COOLDOWN IS IN PROGRESS AND A VOID IN THE REACTOR VESSEL UPPER HEAD IS EXPECTED, THIS PROCEDURE SHOULD NOT BE PERFORMED.

1.	CHECK HHSI Pumps - ALL STOPPED	RETURN To Procedure and Step in effect.

A. PURPOSE

This procedure provides actions to continue plant cooldown and depressurization to cold shutdown with no accident in progress, under conditions that allow for the potential formation of a void in the upper head region with a vessel level system available to monitor void growth.

B. ENTRY CONDITIONS

This procedure is entered from:

1. ES-0.2, NATURAL CIRCULATION COOLDOWN, after completing the first eleven steps.
2. ES-0.2, NATURAL CIRCULATION COOLDOWN, WHEN steam voids have formed in the Reactor vessel and depressurization must continue.
    - Procedure Section, Step 15

      OR

    - ATTACHMENT 1, COOLDOWN WITHOUT CRDM FANS, Step 7

      OR

    - ATTACHMENT 1, COOLDOWN WITHOUT CRDM FANS, Step 16
3. ES-0.4, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), Step 1, WHEN RVLIS is available.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
WS # 10	Group #	1	
	K/A #	000038EA1.16	
OK	Importance Rating	4.4	

Ability to operate and monitor SG atmospheric relief valve and secondary PORV controllers and indicators as they apply to a SGTR

Proposed Question: Common 75

Given the following conditions:

- A Steam generator Tube Rupture has occurred on the 31 SG.
- All equipment is operating as designed.
- 31 SG has been isolated.

The following indications exist:
- 31 SG pressure is 1000 psig and trending UP.
- 31 SG NR level is 55% and trending UP.

Which one of the following describes how pressure will be controlled on 31 SG prior to completion of the RCS depressurization?

A. Automatically at the first SG safety valve setpoint.

B. Manually at the condenser steam dump pressure setpoint.

C. Automatically with the SG atmospheric dump valve controller set at 1040 psig.

D. Manually by performing secondary depressurization to cool down the RCS below initial target temperature.

Proposed Answer:

C. Automatically with the SG atmospheric dump valve controller set at 1040 psig.

Explanation (Optional):

Technical Reference(s): E-3, step 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-EOP30 7 (As available)

Question Source: Bank # INPO 25007
Modified Bank # (Note changes or attach parent)
New

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4,5,7
55.43 5

Comments:

CAUTION

- IF THE TURBINE-DRIVEN AFW PUMP IS THE ONLY AVAILABLE SOURCE OF FEED FLOW, STEAM SUPPLY TO THE TURBINE-DRIVEN AFW PUMP SHOULD BE MAINTAINED FROM AT LEAST ONE SG.
- AT LEAST ONE SG MUST BE MAINTAINED AVAILABLE FOR RCS COOLDOWN.

3. ISOLATE Flow From Ruptured SG(s):

a. ADJUST ruptured SG(s) atmospheric controller to maintain 1040 psig

b. CHECK ruptured SG(s) atmospheric dump valve(s) - CLOSED

Response Not Obtained:

b. WHEN ruptured SG pressure is less than 1040 psig, THEN PERFORM the following:

1) VERIFY atmospheric dump valve(s) closed.

2) IF NOT, THEN PLACE affected atmospheric dump valve(s) in MANUAL and CLOSE.

3) IF atmospheric dump valve(s) can NOT be closed, THEN DISPATCH NPO to close per SOP-ESP-1, LOCAL OPERATION OF SAFE SHUTDOWN EQUIPMENT.

(STEP 3 CONTINUED ON NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 76	Group #		1
	K/A #	025AA2.07	
OK	Importance Rating		3.7

Ability to determine and interpret pump cavitation as it applies to the Loss of Residual Heat Removal System.

Proposed Question: **"SRO ONLY" 76**

Given the following plant conditions:

- The plant is in CSD following a refueling outage.
- A vacuum has been drawn on the RCS in preparation for vacuum filling of the RCS.
- Vacuum is currently 26"Hg
- RCS level is 62'6"
- RHR flow prior to drawing the vacuum was 2500 gpm.
- RCS temperature is 125°F
- RHR Flow Indicator (FI-640) starts fluctuating from 1500 gpm to 2000 gpm with the 31 RHR pump running.

What has caused the reduction in RHR flow?

A. Letdown flow has been lost.

B. 31 RHR pump is cavitating due to high RCS temperature.

C. RCS inventory is being lost due to lifting of RHR discharge header relief valve733A or 733B due to high discharge pressure.

D. 31 RHR Pump is vortexing due to low RCS inventory from draining below mid-loop.

Proposed Answer:

B. 31 RHR pump is cavitating due to high RCS temperature.

Explanation (Optional):

Technical Reference(s): 3-SOP-RCS-017
3-POP-4.2
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-POP-004 A (As available)

Question Source: Bank # INPO
Modified Bank # 28389 (Note changes or attach parent)
New

Question History: 11/15/2004 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

WS 76

____ 4.5.7.4 INITIATE Aux Spray per 3-SOP-CVCS-002, Charging Seal Water And Letdown Control (both spray valves OPEN to minimize flow to PZR).

4.5.8 ISOLATE RCP #1 Seal leakoff stop valves by CLOSING the following:

- ____ CH-AOV-261A 31 RCP Seal Return Isolation Valve
- ____ CH-AOV-261B 32 RCP Seal Return Isolation Valve
- ____ CH-AOV-261C 33 RCP Seal Return Isolation Valve
- ____ CH-AOV-261D 34 RCP Seal Return Isolation Valve

NOTE

- RCS Temperature should be maintained near the temperature used to determine target RCS vacuum until RCS Level is greater than 66' (See P&L 2.3.1)
- IF achievable vacuum is less than 20" Hg
  THEN vacuum SHALL be broken only by steam bubble in PRZR (e.g. 21" Hg)
- Attachment 9, Vacuum Pump Skid Diagram provides clarification for operation of the Vacuum Pump.
- Vacuum SHALL NOT be drawn on the RCS until the Reactor Head has been fully tensioned and all Cono Seals are installed.
- CTOs may be applied to the vacuum pump, to ensure vacuum pump power and seal water are maintained during the vacuum evolution.
  - Temporary Power Supply
  - Seal Water

4.5.9 RCS Evacuation

____ 4.5.9.1 DETERMINE MAXIMUM RCS vacuum using Attachment 1, Maximum Allowable Vacuum.

____ 4.5.9.2 DETERMINE target RCS Vacuum between 20"Hg and maximum determined in previous step (typically 23").

____ 4.5.9.3 DETERMINE RCS Evacuation Time using Attachment 2, Vacuum Refill Performance, Figure RCS-17-2 (page 2 of 4) OR Figure RCS-17-3 (page 3 of 4) for PRT not drained.

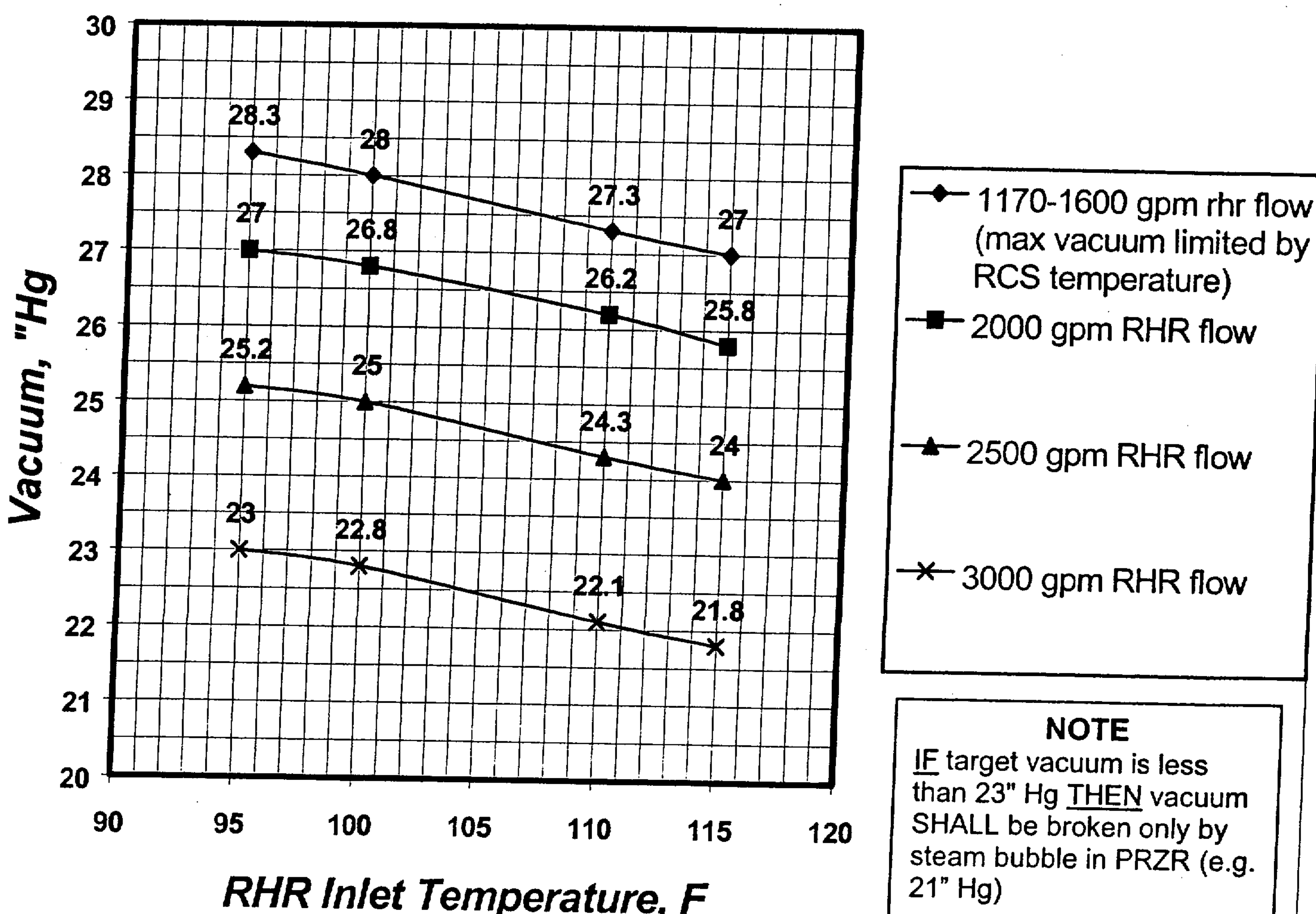
2.3 Temperature Related Precautions and Limitations

2.3.1 RCS temperature SHALL be maintained in accordance with the following table in order to ensure adequate NPSH to the RHR pump while the RCS is under vacuum conditions. (Graph in Attachment 1, Maximum Allowable Vacuum provides pressure and flow guidance)

RCS Level	Allowable RCS Temperature
Less than 66'	100°F to 115°F (Stable)
66' to 67'	UP TO 120°F
67' to 68'	UP TO 122°F
68' to 69'	UP TO 124°F
69' to 70'	UP TO 126°F
70' to 71'	UP TO 128°F
71' to 72'	UP TO 130°F
72' to 73'	UP TO 132°F
73' to 74'	UP TO 134°F
74' to 75'	UP TO 136°F
75' to 76'	UP TO 138°F
76" to 77'	UP TO 140°F
77' to 78' (≅ 10% PRZR level)	UP TO 142°F
>78' - 2°F for every foot of water level added to the RCS.	

2.3.2 WHEN using Ultrasonic level device for RCS level THEN Makeup water temperature should be within 20°F of RCS temperature to minimize density effects on the Ultrasonic Level device.

Attachment 1
Maximum Allowable Vacuum
(Attachment Page 1 of 1)

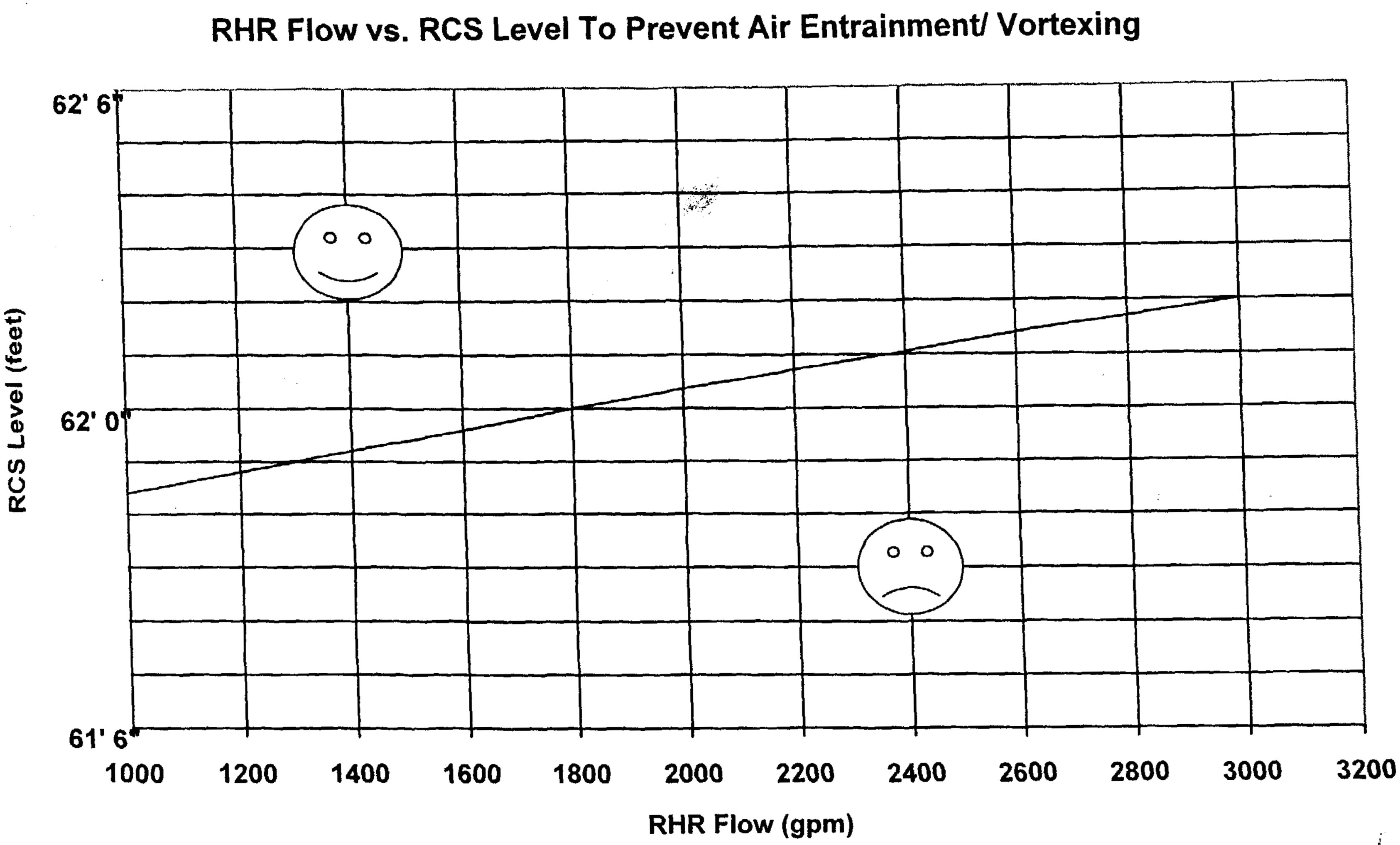


NOTE
IF target vacuum is less than 23" Hg THEN vacuum SHALL be broken only by steam bubble in PRZR (e.g. 21" Hg)

ATTACHMENT 10

RHR FLOW VS. RCS LEVEL TO PREVENT AIR ENTRAINMENT/VORTEXING

(Page 1 of 1)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 77	Group #		1
	K/A #	W/E12EA2.1	
OK	Importance Rating		4.0

Ability to determine and interpret the facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to the Uncontrolled Depressurization of all Steam Generators

Proposed Question: **"SRO ONLY" 77**

Given the following plant conditions:

- The Unit has sustained a main steam line break affecting all 4 SGs.
- The crew is currently performing ECA 2.1, Uncontrolled Depressurization Of All Steam Generators.
- The crew has throttled AFW flow to 100 gpm to each SG to minimize the RCS cooldown. Safety Injection Termination Criteria is NOT met.

SG	Level	Pressure	TREND
31 SG	19% WR	320 psig	SLOWLY DECREASING
32 SG	18% WR	310 psig	SLOWLY DECREASING
33 SG	26% WR	380 psig	SLOWLY INCREASING
34 SG	18% WR	310 psig	SLOWLY DECREASING

Which one of the following describes the required action and the reason for the action?

A. Continue with ECA 2.1, Uncontrolled Depressurization Of All Steam Generators, because Safety Injection termination is not complete.

B. Transition to FR-H.1, Loss Of Secondary Heat Sink because there is a RED condition on the Heat Sink Status Tree.

C. Transition to E-2, Faulted Steam Generator Isolation because there is an intact SG available.

D. Transition to E-3, Steam Generator Tube Rupture because there is an unexplained increase in SG level.

Proposed Answer:

C. Transition to E-2, Faulted Steam Generator Isolation because there is an intact SG available.

Explanation (Optional):

Technical Reference(s): ______________ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # INPO 25026
Modified Bank # ______________ (Note changes or attach parent)
New ______________

Question History: 12/1/2002 Beaver Valley 1

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

1. INTRODUCTION

Guideline ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, provides procedural guidance to recover from an event where all steam generators are depressurizing in an uncontrolled manner. Due to the low probability of an uncontrolled depressurization of all steam generators occurring, additional accidents (e.g., subsequent SGTRs) were not addressed in the development of guideline ECA-2.1.

This guideline is entered from E-2, FAULTED STEAM GENERATOR ISOLATION, Step 2, when an uncontrolled depressurization of all steam generators occurs. Potential initiating events for this contingency could include steamline breaks, stuck open relief or safety valves, or any combination of conditions that would affect all steam generators. Failure of all main steamline isolation valves in conjunction with a single break or stuck open valve could also lead to an uncontrolled depressurization of all SGs in plants that do not have SG non-return valves. Guideline ECA-2.1 is always entered at Step 1.

Guideline ECA-2.1 is exited whenever any steam generator pressure boundary is reestablished as indicated by an increase in the associated steam generator pressure indication. In this case, the operator transfers to guideline E-2 for further recovery actions. It is noted that this transition only applies when the SI termination criteria are not yet met or after SI termination has already been completed in guideline ECA-2.1. Guideline ECA-2.1 is also exited at Steps 6, 7, 9, or 42. Step 6 directs a transition to E-3, STEAM GENERATOR TUBE RUPTURE, if abnormal radiation exists in a steam generator. Step 7 directs a transition to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if RCS pressure is less than the low-head SI pump shutoff head, for further recovery actions. Step 9 directs a transition to ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, if RWST inventory is low. Step 43 is the final step of the guideline. A decision at this point is made by the plant engineering staff concerning future actions with the plant.

FOLDOUT PAGE

1. SI REINITIATION CRITERIA:

IF EITHER condition listed below occurs, THEN manually START SI pumps:

- PRZR level - CAN NOT BE MAINTAINED GREATER THAN 14% [32%]
- RCS subcooling based on qualified core exit TCs - LESS THAN 40°F [SEE TABLE BELOW]

RCS PRESSURE	RCS SUBCOOLING
>1900 psig	[63°F]
>1000 psig	[78°F]
≤1000 psig	[112°F]

2. E-2 TRANSITION CRITERIA:

IF any SG pressure increases at any time (EXCEPT while performing SI termination in ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, Procedure Section, Step 10, through Procedure Section, Step 19), THEN GO To E-2, FAULTED STEAM GENERATOR ISOLATION.

3. E-3 TRANSITION CRITERIA:

- Manually start SI pumps as required, AND Go To E-3, STEAM GENERATOR TUBE RUPTURE if any SG level increases in an uncontrolled manner or any SG has abnormal radiation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 78	Group #		1
	K/A #	055G2.4.30	
OK	Importance Rating		3.6

Knowledge of which events related to system operations/status should be reported to outside agencies

Proposed Question: **"SRO ONLY" 78**

A Loss of Offsite and Onsite power occurred.
The following conditions exist on Unit 3:

- A loss of all AC power occurred 20 minutes ago
- The Emergency Director has classified the event in progress as a Site Emergency
- All State and NRC initial notifications have been made as required
- Maintenance now estimates 5 hours to restore AC power to any 480V bus
- The Emergency Director has upgraded the classification to a General Emergency
- The time now is 01:15

The State of New York must be notified of this change in emergency plan classification NO LATER THAN:

A. 01:30

B. 02:00

C. 02:15

D. 02:30

Proposed Answer:
A. 01:30

Explanation (Optional):

Technical Reference(s): IP-EP-130 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: IOLP-ILO-ERT004 (As available)

Question Source: Bank # INPO 21488
Modified Bank # (Note changes or attach parent)
New

Question History: 7/17/2002 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

5.4 Alert / SAE / GE Upgrade/Update Notifications – CCR/EOF Communicator

5.4.1 Upgrade/Update notifications are made for EAL upgrades and for periodic updates during an Alert, Site Area Emergency (SAE) or General Emergency (GE).

5.4.2 Use an Upgrade/Update Notification Alert/SAE/GE Checklist, (IP-EP-115 Form EP-5) to make and document the emergency classification upgrade or update notifications.

5.4.3 Obtain the completed Radiological Emergency Data Form Part I (IP-EP-115 Form EP-1) from the Shift Manager/Emergency Director AND notify State and Counties within 15 minutes of any emergency classification change or approximately every 30 minutes otherwise. Time intervals may be lengthened with concurrence of offsite agencies.

6.0 INTERFACES

6.1 SOP-CG-7-1, "Notification During Nuclear Emergency Involving IP No. 2"

6.2 IP-EP-115, "Emergency Plan Forms"

6.3 IP-EP-250, "Emergency Operations Facility"

6.4 IP-EP-210, "Control Room"

7.0 RECORDS

All Logs, Completed Forms and other records generated during an actual emergency shall be considered quality records and maintained for the life of the plant.

8.0 REQUIREMENTS AND COMMITTMENTS

NONE

9.0 ATTACHMENTS

9.1 Local Government Radio System Locations & Call Letters

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 79	Group #		1
	K/A #	057A2.19	
	Importance Rating		4.3

Ability to determine and interpret plant automatic actions that will occur on a loss of a vital ac electrical instrument bus as they apply to the loss of a vital instrument bus

Proposed Question: **"SRO ONLY" 79**

Given the following plant conditions:

- Unit 3 is at 100% power
- "B" Steam Flow and Feed Flow are in control
- PRZR pressure channel 1 is in control and 2 is in alarm positions
- PRZR level channel 2 is in control and 1 is in alarm positions
- All systems are in their normal full power lineup
- A loss of 31 Instrument Bus occurs

Which of the below describes the plant response with no operator actions?

A. Letdown isolates, all PRZR heaters de-energize, 31 and 33 Main feed Reg valves go open.

B. Letdown isolates, all PRZR heaters de-energize, ALL four Main feed Reg valves remain at their initial position.

C. Letdown remains in service, all PRZR heaters remain in their initial condition, 31 and 33 Main feed Reg valves go closed.

D. Letdown remains in service, all PRZR heaters remain in their initial condition, ALL four Main feed Reg valves remain at their initial position.

Proposed Answer:

B. Letdown isolates, all PRZR heaters de-energize, ALL four Main feed Reg valves remain at their initial position.

Explanation (Optional):

LCV-459 closes isolating L/D and PRZR heaters de-energize due to low PRZR level and MFW Reg valves have SF and FF matched (0 lbm/hr)

Technical Reference(s): 3-AOP-IB-1, step 4.4, Attachment 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

LJS79

Attachment 5
Loads on 31and 31A Instrument Buses

Ckt	31A Instrument Bus Load	Comment
1	RPS - Analog - Ch. II (Rack A8)	
1A	Rad. Mon. Control Panel PAB Microprocessor Rack (RG 1.97) & Sampling Assy Flow Meter & Temp. Control - Power Supply	Ckt. 1A mounted above Ckt. 1 amplifier Ch. II
2	Cont. Parameters Recorder - Ch. II (Cab. JO2)	
2A	Rad. Mon. Control Panel PAB Microprocessor Rack (RG 1.97) & Sampling Assy Flow Meter & Temp. Control:	Ckt. 2A mounted above Ckt. 2 amplifier Ch. II
	R-56 (Sewage Treatment)	
	R-67 (41' PAB)	
	R-68 (15' PAB)	
	R-69 (55' Pipe Pen)	
	R-70 (80' PAB)	
3	X-Core Neutron Detector N-39 (Loc. 90°)	
4	RCS - Analog - Ch. II (Rack A7)	
5	Spare	
6	CFMS Iso. (Cab. II)	
8	QSPDS Train A	
	QSPDS Train A	
10	MDFP 31 & Lovejoy Speed Controller System	Panel FAF
11	CVCS Aux. (Rack C9)	Panel SAF
12	RVLIS Train A	
13	PAB & Trench Temp. Monitoring Cabinet	
14	Rad. Mon. Rack	
	PAB Microprocessor Rack	
	R-26 (Cont. Bldg. DHRRM)	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 80	Group #		1
	K/A #	058G2.1.30	
OK	Importance Rating		3.4

Ability to locate and operate components, including local controls.

Proposed Question: **"SRO ONLY" 80**

Given the following conditions:

- The plant is at 100% power.
- 34 and 36 Service Water Pumps (SWPs) are in service supplying the essential header.
- 125 VDC control power to the 34 SWP is lost.

Which ONE (1) of the following describes the effect on the operation of 34 SWP and what actions, if any, would be required should 34 SWP need to be secured?

A. Breaker indication in CCR is lost, CCR breaker control is lost, pump remains running – Remove control power fuses, press trip plate on cubicle door to secure the pump.

B. Breaker indication in CCR is available, CCR breaker control is lost, pump will remain running – Remove control power fuses, press trip plate on front of breaker inside cubicle to secure the pump.

C. Breaker indication is available, CCR breaker control is lost, pump will trip – no further action required to secure the pump.

D. Breaker indication in CCR is lost, CCR breaker control is lost, pump will trip – no further action required to secure the pump.

Proposed Answer:

A. Breaker indication in CCR is lost, CCR breaker control is lost, pump remains running – Remove control power fuses, press trip plate on cubicle door to secure the pump

Explanation (Optional):

Technical Reference(s): 3-AOP-DC-1, Attachment 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Valves in the following step have valve stem locking devices (screws).

4.10 **IAAT** CRS/SM desires to throttle charging pump recirc valve(s) to aid inventory control,
THEN dispatch an operator to throttle the following:

	PUMP	VALVE
	31	CH-407 (31 CHARGING PUMP RECIRCULATION ISOLATION)
	32	CH-408 (32 CHARGING PUMP RECIRCULATION ISOLATION)
	33	CH-409 (33 CHARGING PUMP RECIRCULATION ISOLATION)

4.11 __**INITIATE** investigation of fault.

4.12 **PERFORM** Attachment 12 (Local Operation of 6900V and 480V Breakers) (Page 55) as necessary to operate the following:

__6900V breakers on Buses 1, 2, or 5

__480V breakers on Bus 5A

4.13 __**IAAT** operation of 52/UT3ST6 or 52/UT4ST6 is required,
THEN PERFORM applicable section of SOP-EL-005 (Operation of On-Site Power Sources) to operate breakers.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.14 __Is 31 DC Power Panel de-energized?	__ **WHEN** affected DC Distribution Panel has been re-energized, **THEN GO TO** Step 4.21.

Attachment 12
Local Operation of 6900V and 480V Breakers

Operating 480V Breakers	
Task	**Steps**
Trip load, feed or tie breaker	1. Remove breaker control power fuses. 2. Press trip plate on cubicle door or trip plate on front of breaker (inside cubicle).
Close load, feed or tie breaker (See notes below)	1. Remove breaker control power fuses. 2. Charge spring until spring status plate indicates "Spring Charged". 3. Press Push-To-Close button.

- Breakers 52/312, 52/313, and 312-313 tie have a key interlock system that prevents closing of all three breakers at the same time. Normally, two supply breakers are closed and tie breaker is locked in open position by interlock. To close tie breaker, either supply breaker must first be locked open (instructions are inside cubicle door).
- Feed breakers for motor-driven fire pump (Buses 312 and 5A) and the associated manual transfer switch (CB 15' el., Northwest corner) have a Kirk key interlock system that prevents operation of the transfer switch under energized conditions. To operate the transfer switch, both feed breakers must first be locked open (instructions are inside cubicle door for each feed breaker).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 81	Group #		1
	K/A #	W/E05EA2.1	
OK	Importance Rating		4.4

Ability to determine and interpret facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question: **"SRO ONLY" 81**

Given the following:

- Unit 3 has had a loss of both Feedwater Pumps from 100% power.
- SG LOW level annunciators alarm and the Reactor failed to trip.
- Actions of FR-S.1, Response to Nuclear Power Generation / ATWS are being performed.
- All AFW pumps failed to start and cannot be started.
- Reactor Power has just been verified to be < 5%, with a negative start up rate.

Which one of the following procedures should the SRO transition to?

A. Immediately enter FR-H.1, Response to Loss of Secondary Heat Sink.

B. Complete all actions in FR-S.1, then transition to FR-H.1, Response to Loss of Secondary Heat Sink.

C. Re-enter E-0, Reactor Trip or Safety Injection at the beginning and transition to ES-0.1 Reactor Trip Response when directed by E-0.

D. Re-enter E-0, Reactor Trip or Safety Injection at step 1, complete immediate operator actions and then transition to FR-H.1, Response to Loss of Secondary Heat Sink.

Proposed Answer:

B. Complete all actions in FR-S.1, then transition to FR-H.1, Response to Loss of Secondary Heat Sink.

Explanation (Optional):

Technical Reference(s): OAP-12 (Attach if not previously provided)
FR-S.1

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank # INPO
Modified Bank # 25815 (Note changes or attach parent)
New

Question History: 3/14/2003 Surry 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43

Comments:

WS 81

4.3.9 IF during the performance of any RED condition Function Restoration Procedure (FRP), a RED condition of higher priority occurs, THEN the higher priority RED condition should be addressed first, AND the lower RED condition FRP suspended.

4.3.10 IF any ORANGE terminus is encountered, THEN the operator is expected to monitor ALL of the remaining trees, AND THEN, IF NO RED is encountered, SUSPEND any E-Set procedure in progress AND transition to the FRP required by the ORANGE terminus.

4.3.11 IF during performance of an ORANGE condition FRP, any RED condition or higher priority ORANGE condition arises, THEN the RED or higher priority ORANGE condition is addressed first, AND the original ORANGE condition FRP suspended. IF a FRP specifically states that higher priority condition should NOT be addressed, THEN this requirement does NOT apply.

4.3.12 Once a FRP is entered due to a RED or ORANGE condition, that FRP is performed to completion, unless that FRP is preempted by a higher priority condition.

4.3.13 It is expected that the actions in the FRP will clear the RED or ORANGE condition before all operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the return to "procedure and step in effect".

4.3.14 Status Tree monitoring should be CONTINUOUS IF any ORANGE or RED condition is found to exist. IF no condition more serious than YELLOW is encountered, THEN monitoring frequency may be reduced to 10-20 minutes, UNLESS some significant change in plant status occurs.

4.3.15 A YELLOW terminus does not require immediate operator attention. Frequently it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the YELLOW status might provide an early indication of a developing RED or ORANGE condition. Following FRP implementation, a YELLOW might indicate a residual off-normal condition. The operator is allowed to decide whether or not to implement any YELLOW condition FRP.

4.3.16 The operator should be familiar with all instrumentation used in the status trees. The only unique application is the use of core exit thermocouples in the Core Cooling Status Tree, where the implementation of FRPs as a result of core exit temperature should be based on at least five (5) thermocouples reading greater than the action temperature level. Therefore, FRP implementation is based on the fifth highest core exit thermocouple reading, when used in the status tree.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	CHECK Core Exit TCs - LESS THAN 1200°F	IF core exit temperatures are greater than 1200°F AND increasing, THEN GO To SACRG-1, Severe Accident Control Room Guideline Initial Response.
15.	VERIFY Reactor Subcritical:	
	a. CHECK Reactor Power and SUR: • Power range channels - LESS THAN 5% • Intermediate range channels - Zero or Negative startup rate	a. PERFORM the following: 1) CONTINUE to borate. 2) IF boration is NOT available, THEN ALLOW RCS to heat up. 3) PERFORM actions of other Function Restoration Procedures in effect which do NOT cool down OR otherwise add positive reactivity to the core. 4) RETURN To Step 4, Page 5.
	b. CHECK all rods - LESS THAN 20 STEPS	b. CONTINUE to insert rods manually until all rods are LESS THAN 20 STEPS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 82	Group #		2
	K/A #	00005AA2.03	
OK	Importance Rating		4.4

Ability to determine and interpret the required actions if more than one rod is stuck or inoperable as they apply to the inoperable / Stuck Rod

Proposed Question: **"SRO ONLY" 82**

Given the following plant conditions:

- The unit is at 88% power
- Control Bank "D" is at 210 steps
- A single rod misalignment has been discovered (14 steps out of alignment).
- Core peaking factors have been determined to be within limits.
- Attempts to realign the rod have failed and the rod has been determined to be mechanically stuck.

Subsequently, a second control rod falls out of alignment criteria (13 steps out of alignment). Attempts to realign this rod have also failed and this rod has been determined to be mechanically stuck. Core peaking factors are within limits with both rods misaligned.

Using the attached Technical Specification reference, which ONE of the following indicates the course of action the operating crew should now take?

A. Reduce reactor power to less than 50% because rod misalignment criteria does not apply at power levels less than 50%

B. Power can be maintained at 88% since core peaking factors are within the limit

C. Commence a plant shutdown because the shutdown margin requirements are no longer met.

D. Reduce reactor power to less than 85% in order to increase the available rod misalignment allowance from 12 to 24 steps.

Proposed Answer:

C. Commence the standard shutdown sequence because the shutdown margin requirements are no longer met

Explanation (Optional):

Technical Reference(s): ITS 3.1.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: ITS 3.1.4

Learning Objective: (As available)

Question Source: Bank # INPO 21615
Modified Bank # (Note changes or attach parent)
New

Question History: 9/6/2002 Kewaunee, Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with rod group alignment limits as follows:

a. When THERMAL POWER is > 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within the limits specified in Table 3.1.4-1 for the group step counter demand position; and

b. When THERMAL POWER is ≤ 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within 24 steps.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	<u>OR</u>	
	B.2.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	<u>AND</u>	
	B.2.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>	
	B.2.4 Perform SR 3.2.1.1.	72 hours
	<u>AND</u>	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B (continued)	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR. OR D.1.2 Initiate boration to restore required SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	------------------ NOTE ---------------------- Not required to be met for individual control rods until 1 hour after completion of control rod movement. --- Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core $\geq$ 10 steps in one direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

Table 3.1.4-1

Maximum Permissible Rod Misalignment
(Indicated Rod Position minus Group Step Counter Demand Position)
When > 85 % RTP

Step Counter Demand Position (steps)	Maximum Permissible Deviations (IRPI Position minus Step Counter Demand Position) (steps)
≤ 212	≥ -12 and ≤ +12
213 to 225	≥ -12 and ≤ +17
226	≥ -13 and ≤ +17
227	≥ -14 and ≤ +17
228	≥ -15 and ≤ +17
229	≥ -16 and ≤ +17
≥ 230	≥ -17 and ≤ +17

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 83	Group #		2
	K/A #	000036G2.2.27	
OK	Importance Rating		3.5

Knowledge of the refueling process

Proposed Question: **"SRO ONLY" 83**

Given the following:

- Refueling operations are in progress
- Irradiated fuel is being moved in the Manipulator Crane from the core to the Containment upender for transfer to the spent fuel pool.
- Decreasing Spent Fuel Pool (SFP) water level has been reported.

Identify the responsibility of the SRO in containment assigned to the fuel shuffle during this event.

A. Direct personnel in FSB to isolate the SFP from the cavity per 3-AOP-FH-1.

B. Locate the Manipulator Crane to the south end of the Reactor Cavity.

C. Evacuate ALL personnel from containment.

D. Evacuate ALL personnel from the Fuel Storage Building.

Proposed Answer:

A. Direct personnel in FSB to isolate the SFP from the cavity per 3-AOP-FH-1.

Explanation (Optional):

Technical Reference(s): 3-AOP-SF-1 (Attach if not previously

provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 6

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Unit Status

A loss of level has occurred in SFP or Refueling Cavity.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.23. __ Is fuel transfer tube gate valve open?	1. __ **IF** SFP level is decreasing, **THEN GO TO** Step 4.33. 2. __ **GO TO** Step 4.39.
4.24. __ Notify personnel in FSB to perform Attachment 1 (FSB Actions for Loss of Level) (Page 41).	
4.25. __ Notify personnel in VC to perform Attachment 2 (VC Actions for Loss of Level) (Page 89).	
4.26. __ Evacuate all nonessential personnel from FSB and VC.	
4.27. __ Notify HP to report to FSB.	
4.28. __ Dispatch at least one operator to determine cause of level decrease.	
4.29. __ **WHEN** notified of SFP to refueling cavity isolation status, **THEN** continue in this procedure.	
4.30. __ Is refueling cavity isolated from SFP?	__ **GO TO** Step 4.42.
4.31. __ Is level decreasing in SFP?	__ **GO TO** Step 4.42.

NOTE

All further actions for an SFP level decrease are controlled locally by Attachment 1.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.32. __ **RETURN** to procedure and step in effect.	

••• END •••

Attachment 1
FSB Actions for Loss of Level

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.1. __ Is fuel transfer canal gate closed and latched?	1.__ Close and latch fuel transfer canal gate. 2.__ **IF** fuel transfer canal gate **CANNOT** be closed and latched, **THEN GO TO** Step 1.4.
1.2. __ Is fuel transfer canal gate seal inflated?	1.__ Inflate fuel transfer canal gate seal to 5 - 25 psig. 2.__ **IF** fuel transfer canal gate has been successfully inflated, **THEN** notify CCR that SFP is isolated from refueling cavity via fuel transfer canal gate. 3.__ **GO TO** Step 1.4.
1.3. __ Notify CCR that SFP is isolated from refueling cavity via fuel transfer canal gate.	
1.4. __ Is fuel transfer car in refueling cavity?	Move fuel transfer car to refueling cavity using one of the following methods: __ Electrically __ Using manual hand-wheel on Transfer Motor __ Using emergency withdrawal cable
1.5. __ Is fuel transfer tube gate valve closed?	1.__ Close fuel transfer tube gate valve. 2.__ **IF** fuel transfer tube gate valve is closed, **THEN** notify CCR that SFP is isolated from refueling cavity via fuel transfer tube gate valve. 3.__ **GO TO** Step 1.7.
1.6. __ Notify CCR that SFP is isolated from refueling cavity via fuel transfer tube gate valve.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 84	Group #		2
	K/A #	W/E06G2.4.4	
OK	Importance Rating		4.3

Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: **"SRO ONLY" 84**

The Reactor has tripped with a loss of offsite power. SI has actuated. The crew is performing actions in E-0, Reactor Trip or Safety Injection. Given the following conditions:

- RCS pressure 1700 psig and trending up
- 31, 32, 34 SG pressures = 1015 psig stable
- 33 SG pressure = 700 psig and trending down
- CETs 750°F and trending up
- SG Narrow Range level off scale Low
- Maximum available AFW flow of approximately 75 gpm to each SG
- PRZR level 15% and trending down
- CNMT pressure 5 psig and trending up
- Power is 2% in the PR and IR SUR is slightly negative
- RVLIS level 45%

Which ONE of the following describes the first procedure transition from E-0?

A. E-2, Faulted Steam Generator Isolation

B. FR-S.1, Response to Reactor Restart/ATWS

C. FR-C.1, Response to Inadequate Core Cooling

D. FR-H.1, Response to Loss of Secondary Heat Sink

Proposed Answer:

D. FR-H.1, Response to Loss of Secondary Heat Sink

Explanation (Optional):

Technical Reference(s): E-0, step 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # INPO 28255

Modified Bank # X (Note changes or attach parent)

New

Question History: 11/1/2004 Ginna 1

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 2

Comments:

WS 84

Number:	Title:	Revision Number:
E-0	REACTOR TRIP OR SAFETY INJECTION	21

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	<u>CHECK AFW Status:</u>	
	a. VERIFY total AFW flow - GREATER THAN 365 GPM	a. PERFORM the following: 1) Manually START available pump(s). 2) ALIGN valves as required. 3) <u>IF</u> cutback controller is malfunctioning, <u>THEN</u> ATTEMPT manual control. 4) <u>IF</u> all SG NR levels are less than 9% [14%] <u>AND</u> total AFW flow can <u>NOT</u> be maintained greater than 365 gpm, <u>THEN</u> GO To FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.
	b. CONTROL feed flow to maintain SG NR levels between 9% [14%] and 50%	

<u>CAUTION</u>

STARTING OF EQUIPMENT MUST BE COORDINATED WITH ALL CONTROL ROOM OPERATORS TO ENSURE THAT TWO COMPONENTS ARE <u>NOT</u> STARTED AT THE SAME TIME ON THE SAME POWER SUPPLY.

* *

6. * <u>DIRECT BOP Operator to PERFORM RO-1, BOP OPERATOR ACTIONS DURING USE OF EOPS</u> *

* *

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
WS # 85	Group #		2
	K/A #	W/E15G2.1.7	
OK	Importance Rating		4.4

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation

Proposed Question: **"SRO ONLY" 85**

Unit 3 experienced a Safety Injection and Containment Spray actuation due to a large break LOCA. E 1, Loss of Reactor or Secondary Coolant, is being performed following a transition from E 0, Reactor Trip or Safety Injection. The STA has just made his initial scan of the Status Trees. The following conditions exist

- Pressurizer level is 0%
- Cnmt pressure is 2.8 psig
- Containment Rad Monitors, R-25 and R-26, have just gone into ALARM
- Containment Sump Level is 51 ft.

Which of the following procedures must be entered to address the above conditions?

A. FR I.2, Response to Low Pressurizer Level

B. FR Z.3, Response to High Containment Radiation Level

C. FR Z.1, Response to High Containment Pressure

D. FR Z.2, Response to Containment Flooding

Proposed Answer:

D. FR Z.2, Response to Containment Flooding

Explanation (Optional):

Technical Reference(s): F-0.5, F-0.6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:

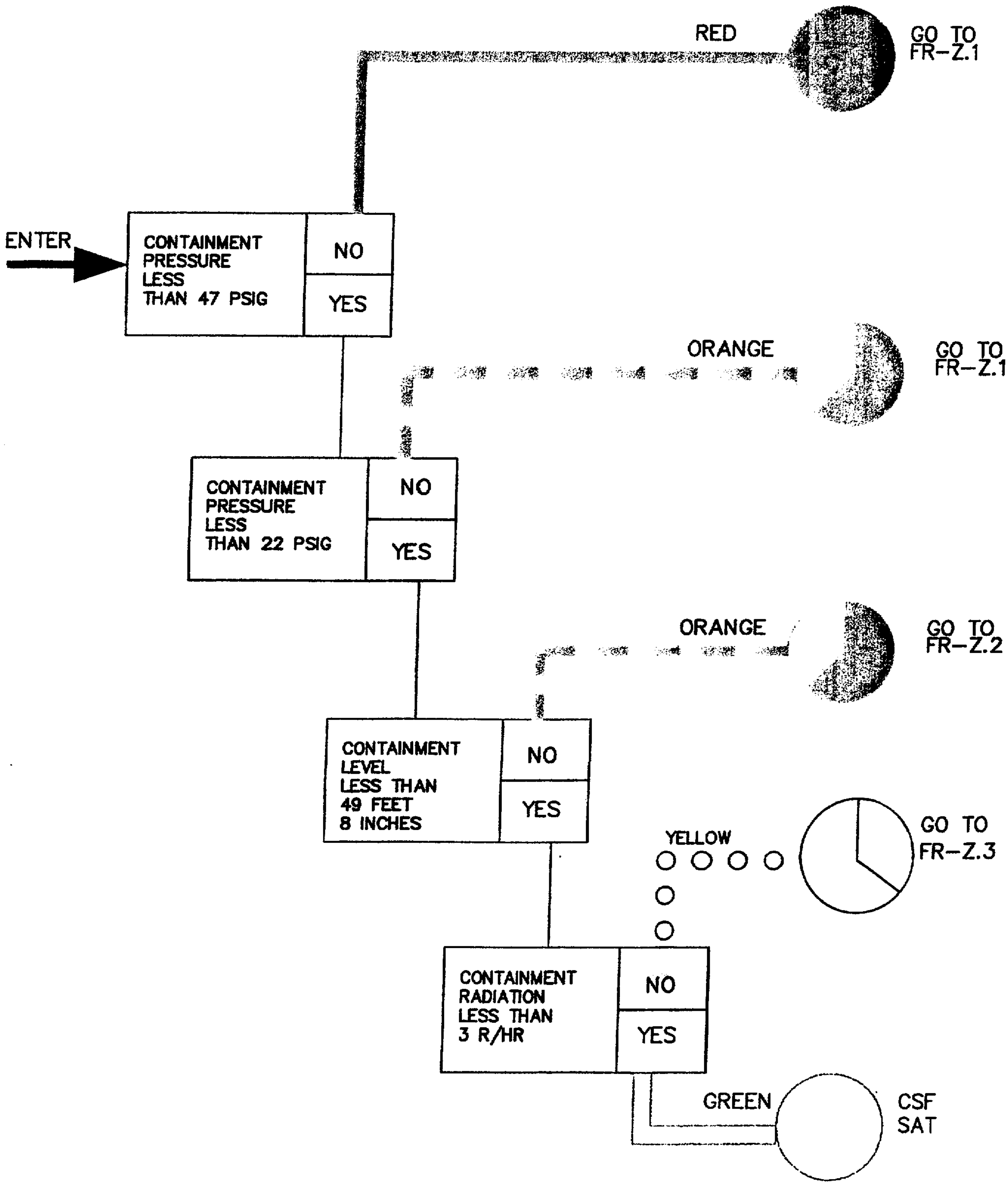
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

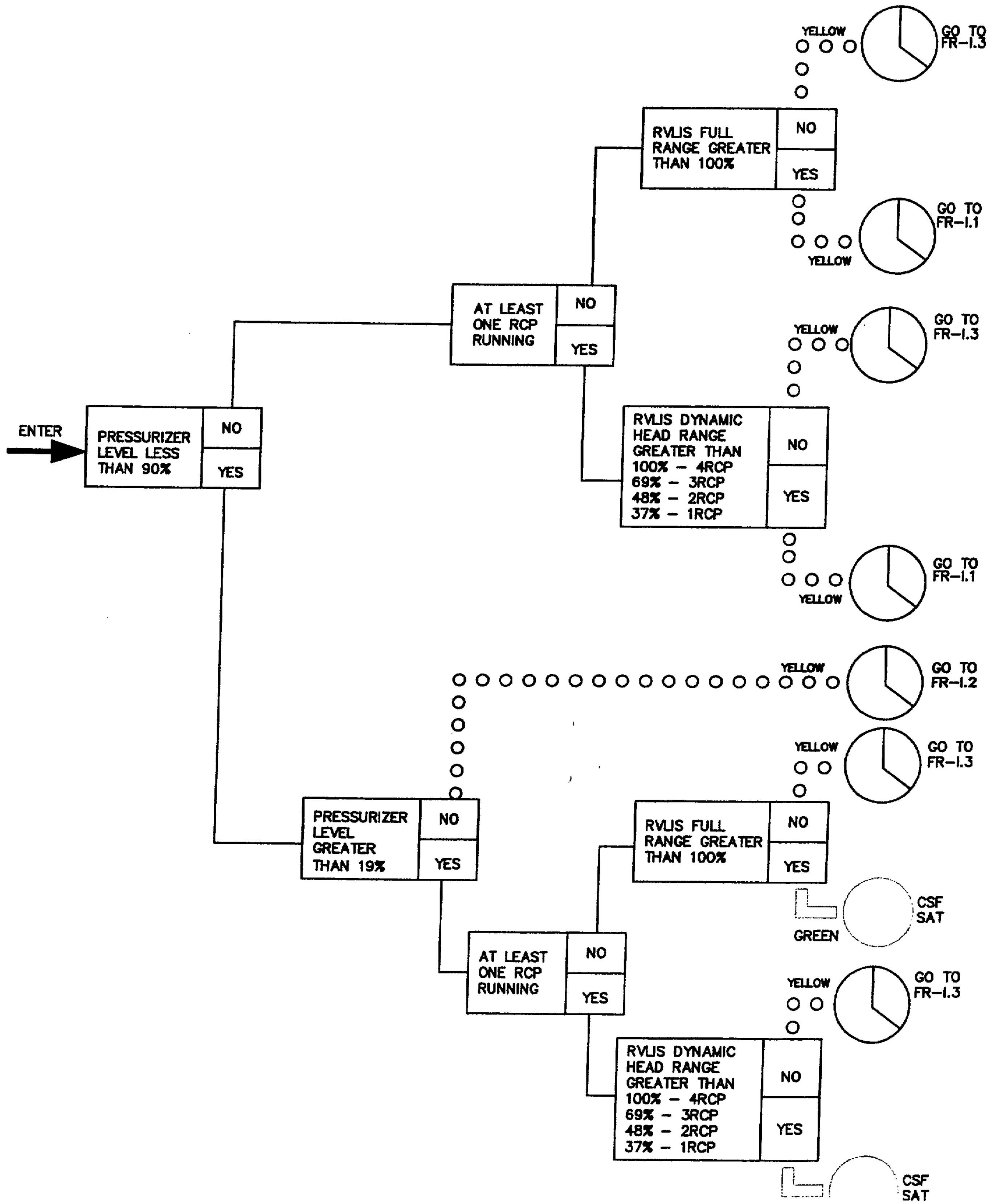
Comments:

W585

Number:	Title:	Revision
F-0.5	CONTAINMENT	8



Number:	Title:	Revision
F-0.6	INVENTORY	10



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 86	Group #		1
	K/A #	004A2.32	
OK	Importance Rating		3.9

Ability to predict the impacts of expected reactivity changes after valving in a new mixed-bed demineralizer that has not been pre-borated and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations

Proposed Question: **"SRO ONLY" 86**

The following plant conditions exist:

- Unit 3 is in a normal full power lineup at MOL
- A fresh CVCS purification demineralizer has been placed in service.
- The resin has NOT been boron saturated.

Which of the following is the expected plant response, with no operator action and what operator actions should be taken to mitigate this event?

A. Power level will go up, commence emergency boration using MOV-333 to maintain power $\leq$ 100% and T_{AVG} at program value.

B. Power level will go up, commence a normal boration to maintain power $\leq$ 100% and T_{AVG} at program value.

C. RCS pressure will decrease energize Pressurizer Backup Heaters to maintain RCS pressure at 2235 psig.

D. RCS pressure will increase; trip the Reactor, initiate E-0, Reactor Trip or Safety Injection.

Proposed Answer:

B. Power level will go up, commence a normal boration to maintain power $\leq$ 100% and T_{AVG} at program value.

Explanation (Optional):

Technical Reference(s): SD-3.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # INPO
Modified Bank # 18456 (Note changes or attach parent)
New

Question History: 1/1/2000 San Onofre 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

Replacement - CVCS Demineralizers, contains all the required steps for resin replacement.

2.2.1 Mixed Bed Demineralizers

General purification of the letdown flow takes place in the resin bed of one of the two full flow mixed bed demineralizers. The resin bed performs ion exchange and mechanical filtration of particles greater than 25 microns. The resin bed is lithium hydroxide and the basic ion exchange process releases Li^+ and OH^- ions from the demineralizer effluent. Due to the relatively high concentration of boric acid in the coolant, the BO_3^{-3} present rapidly exchanges with the anion resin forming $H_2BO_3^{-3}$ (di-hydrogen borate) when the resin is initially placed in service. The reduction in boron adds positive reactivity to the RCS. The decrease in boron concentration in the RCS that occurs when a fresh resin bed is placed in service is about 100 ppm at BOL and about 40 ppm at EOL. The OH^- ions released in the effluent will increase the system pH. After the bed reaches equilibrium, impurities exchange with the anion resin to release BO_3^{-3} ions and the mixed beds have little effect on pH. The 66% anion and 33% cation mixture ensures that the effluent will be neutral (pH = 7) because the Li cation has a tendency to create a strong basic solution.

In addition to removing general ionic impurities, the resin bed is designed to reduce the concentration of isotopes in the purification stream by a minimum decontamination factor (ratio of inlet to outlet activity) of 10, with the exception of cesium. Each demineralizer is sized to accommodate a maximum letdown flow of 120 gpm and has a sufficient capacity to reduce RCS activity to refueling concentrations after operating for one core cycle with one percent fuel defects. The mixed bed will remove hydrazine (added for oxygen scavenging at plant startup) and should be bypassed during hydrazine additions to prevent early depletion of the resin bed and unwanted removal of hydrazine.

2.2.2 Cation Bed Demineralizer (Figures 3.0-5 and 6)

The cation bed demineralizer is located downstream of the mixed bed demineralizers and is normally bypassed. It is not a full flow demineralizer and will only accommodate up to 42 gpm (limited to 40 gpm by procedure). The resin is a hydrogen form cation, and is used intermittently to maintain cesium 137 activity concentration in the coolant below 1.0 µci/cc (with one percent fuel defects) and to control the concentration of lithium-7.

$Li^+ + H^+R^-$(cation resin) = Li^+R^-(Li resin)$+H^+$

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 87	Group #		1
	K/A #	012G2.2.25	
OK	Importance Rating		3.7

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits

Proposed Question: **"SRO ONLY" 87**

Given the following plant conditions:

- Unit 3 is at 100% power.
- PT-412B, First Stage Turbine Pressure, has just failed low.

What action is required by Technical Specifications and why?

A. Within 30 minutes verify the P-7 interlock relay is de-energized to ensure the PZR pressure LOW, PZR water level HIGH, RCS flow LOW (2 Loops), RCP breaker open (2 Loops), RCP undervoltage and RCP under frequency Reactor Trips are enabled.

B. Within 1 hour verify the P-7 interlock relay is de-energized to ensure the PZR pressure LOW, PZR water level HIGH, RCS flow LOW (2 Loops), RCP breaker open (2 Loops), RCP undervoltage and RCP under frequency Reactor Trips are enabled.

C. Within 30 minutes verify the P-10 interlock relay is energized to allow ensure the PZR pressure LOW, PZR water level HIGH, RCS flow LOW (2 Loops), RCP breaker open (2 Loops), RCP undervoltage and RCP under frequency Reactor Trips are enabled.

D. Within 1 hour verify the P-10 interlock relay is energized to ensure the PZR pressure LOW, PZR water level HIGH, RCS flow LOW (2 Loops), RCP breaker open (2 Loops), RCP undervoltage and RCP under frequency Reactor Trips are enabled.

Proposed Answer:

B. Within 1 hour verify the P-7 interlock relay is de-energized to ensure the PZR pressure LOW, PZR water level HIGH, RCS flow LOW (2 Loops), RCP breaker open (2 Loops), RCP undervoltage and RCP under frequency Reactor Trips are enabled.

Explanation (Optional):

Technical Reference(s): TS 3.3.1 Condition N and bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 ____

Comments:

WS 8+

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
N. One or more channels inoperable.	N.1	Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>		
	N.2	Be in MODE 2.	7 hours
O. One trip mechanism inoperable for one RTB.	O.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	<u>OR</u>		
	O.2.	Be in MODE 3.	54 hours

Table 3.3.1-1 (page 5 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
15. Turbine Trip-Auto-Stop Oil Pressure	1(h)	3	J	SR 3.3.1.10 SR 3.3.1.14	NA
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	K	SR 3.3.1.14	NA
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(d)	2 trains	M	SR 3.3.1.11 SR 3.3.1.13	NA
b. Low Power Reactor Trips Block, P-7	1	2 trains	N	SR 3.3.1.11 SR 3.3.1.13	NA
c. Power Range Neutron Flux, P-8	1	4	N	SR 3.3.1.11 SR 3.3.1.13	NA
d. Power Range Neutron Flux, P-10	1,2	4	M	SR 3.3.1.11 SR 3.3.1.13	NA
e. Turbine First Stage Pressure, P-7 Input	1	2	N	SR 3.3.1.1 SR 3.3.1.10 SR 3.3.1.13	NA

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(h) Above the P-8 (Power Range Neutron Flux) interlock.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection if required.

The Allowable Value is NA for this function because there is no corresponding analytical limit modeled in the accident analysis. The surveillance acceptance criterion used for this Function is ≥3.1E-11 Amps.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock, is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine First Stage Pressure. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock (i.e., 2 of 4 Power Range channels increasing above the P-10 (Function 17.d) setpoint or 1 of 2 Turbine First Stage Pressure (Function 17.e) setpoint) automatically enables reactor trips on the following Functions:

- Pressurizer Pressure—Low;
- Pressurizer Water Level—High;
- Reactor Coolant Flow—Low (Two Loops);
- RCPs Breaker Open (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock (i.e., 3 of 4 Power Range channels decreasing below the P-10 (Function 17.d) setpoint and 2 of 2 Turbine First Stage Pressure channels decreasing below the Turbine First Stage Pressure (Function 17.e) setpoint) automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure—Low;
- Pressurizer Water Level—High;
- Reactor Coolant Flow—Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs

An Allowable Value is not applicable to the P-7 interlock because it is a logic Function. The P-10 interlock (Function 17.d) governs input from the Power Range instruments and the Turbine First Stage Pressure interlock (Function 17.e) governs input for turbine power.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train (i.e., two trains) of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 88	Group #		1
	K/A #	059A2.07	
OK	Importance Rating		3.3

Ability to predict the impacts of a trip of MFW pump turbine on the MFW and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations

Proposed Question: **"SRO ONLY" 88**

Given the following plant conditions;

- Unit 3 is performing a power ascension and is currently at 77% power
- 31 and 33 Condensate Pumps are in service
- 32 Condensate Pump is secured but available
- Both Heater Drain Tank Pumps are in service
- Both MBFPs are in service

A problem develops with 32 MBFP Thrust Bearing causing the pump to TRIP.

What is the appropriate course of action for the above conditions?

A. Perform the immediate operator actions of 3-AOP-FW-1, Loss of Feedwater, reduce load to approximately 700 MWE, adjust speed on 31 MBFP as necessary to maintain suction pressure >350 psig and discharge pressure <1390 psig and then start 32 Condensate Pump.

B. Commence a rapid load reduction to 500 MWE, perform the immediate operator actions of 3-AOP-FW-1, start 32 Condensate Pump and then adjust speed on 31 MBFP as necessary to maintain suction pressure >350 psig and discharge pressure <1390 psi.

C. Perform the immediate operator actions of 3-AOP-FW-1, trip the Reactor and enter E-0, Reactor Trip or Safety injection.

D. Trip the Reactor and enter E-0.

Proposed Answer:

A. Perform the immediate operator actions of 3-AOP-FW-1, Loss of Feedwater, reduce load to approximately 700 MWE, adjust speed on 31 MBFP as necessary to maintain suction pressure >350 psig and discharge pressure <1390 psig and then start 32 Condensate Pump.

Explanation (Optional):

Technical Reference(s): 3-AOP-FW-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43 5

Comments:

1. PURPOSE

To respond to a reduction of or total loss of feedwater flow due to loss of MBFPs, condensate pumps, heater drain pumps, feedwater valve failure, or feed line break.

2. ENTRY CONDITIONS

Any unanticipated reduction in feedwater flow.

3. IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.1 __ Is any MBFP operating?	1.__ **IF** reactor power is > 4%, **THEN** trip the reactor and **GO TO** E-0. 2.__ **GO TO** Step 4.1.
3.2 __ Are both MBFPs operating?	__ **IF** reactor power is > 80%, **THEN** trip the reactor and **GO TO** E-0.
3.3 __ **GO TO** Step 4.1.	

4. SUBSEQUENT ACTIONS

<table>
<tr><th>ACTION/EXPECTED RESPONSE</th><th>RESPONSE NOT OBTAINED</th></tr>
<tr><td>4.1 GO TO applicable step for transient indicated:
<table>
<tr><th></th><th>TRANSIENT</th><th>STEP</th></tr>
<tr><td></td><td>MBFP trip</td><td>4.2</td></tr>
<tr><td></td><td>Loss of MBFP speed control</td><td>4.15</td></tr>
<tr><td></td><td>Heater drain or condensate pump trip</td><td>4.39</td></tr>
<tr><td></td><td>Valve malfunction in Main FW discharge line</td><td>4.53</td></tr>
<tr><td></td><td>FW line break</td><td>4.92</td></tr>
</table></td><td></td></tr>
<tr><td>4.2 __ Is a MBFP running?</td><td>__ GO TO Step 4.10.</td></tr>
<tr><td>4.3 __ Are steam flow and feed flow matched?</td><td>1.__ INITIATE load reduction as necessary per Attachment 1 (Approximate Unit Load With Various Pump Configurations) (Page 35).
2.__ Adjust running MBFP speed as necessary to accomplish the following:
• Match steam flow and feed flow
• Maintain MBFP suction pressure > 350 psig
• Maintain MBFP discharge pressure < 1390 psig</td></tr>
<tr><td>4.4 __ Are <u>all</u> available condensate pumps running?</td><td>__ Start <u>all</u> available condensate pumps as necessary to maintain MBFP suction > 350 psig.</td></tr>
<tr><td>4.5 __ Are 31 and 33 ABFP running?</td><td>__ Start 31 and 33 ABFPs.</td></tr>
</table>

Attachment 1
Approximate Unit Load With Various Pump Configurations

Page 1 of 1

Main Boiler Feed Pumps(1)	Condensate Pumps	Condensate Booster Pumps (If I/S)(2)	Heater Drain Pumps	Approximate Allowable MWE
2	3	2	2	Full Load
2	2	2	2	900 MWE(3)
2	3	2	1	700 MWE(3)
1	2	2	2	700 MWE(3)
1	2	2	1	550 MWE(3)
2	2	2	1	550 MWE(3)
2	2	2	0	400 MWE(3)

(1) Each MBFP is rated at 15300 gpm at 4740 rpm with a discharge pressure of 970 psig. A single MBFP can supply a maximum flowrate of 18900 gpm calculated on a head curve at 4875 rpm (expected to occur at approximately 700 MWE).

(2) Operation of condensate booster pumps at less than 400 MWE will be dictated by running pump current and total condensate flow. Condensate booster pump running current should be maintained less than 53 amps.

(3) Load will be limited by MBFP suction pressure. Per 3-SOP-FW-001, the suction pressure lower limit is 350 psig during normal operation. At 310 psig, the Standby Condensate Booster Pumps will automatically start (if in AUTO) and CD-AOV-521 (POLISHER VESSELS AND POST FILTERS BYPASS) automatically opens. An automatic low suction pressure cutback actuates at 265 psig to lower MBFP speed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 89	Group #		1
	K/A #	061G2.1.12	
	Importance Rating		4.0

Ability to apply technical specifications for a system

Proposed Question: **"SRO ONLY" 89**

Given the following conditions with Unit 3 operating at 100% power:

- 31 Auxiliary Feedwater (AFW) Pump is out of service for repairs. Repairs will take at least 24 more hours.
- A routine QA Audit of completed surveillance procedures has determined the quarterly surveillance performed on 33 AFW Pump 35 days ago was NOT properly completed.

In accordance with Technical Specifications, which one of the following actions is correct for this situation?

A. Enter T.S. LCO 3.0.3 and IAW T.S. SR 3.0.4, re-perform the surveillance on 33 AFW Pump within 24 hours

B. Enter T.S. LCO 3.0.3 but the required actions can be delayed for 24 hours IAW T.S. SR 3.0.3.

C. Enter T.S. 3.7.5 Condition C, i.e., 2 AFW Trains inoperable, but the required actions can be delayed for 24 hours IAW T.S. SR 3.0.4.

D. Continue T.S 3.7.5 Condition B, i.e., 1 AFW Train inoperable. Re-perform the surveillance on 33 AFW Pump within 24 hours IAW T.S. SR 3.0.3.

Proposed Answer:

D. Continue T.S 3.7.5 Condition B, i.e., 1 AFW Train inoperable. Re-perform the surveillance on 33 AFW Pump within 24 hours IAW T.S. SR 3.0.3.

Explanation (Optional):

Technical Reference(s): T.S. 3.7.5 & SR 3.0.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # INPO 23173
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: 11/4/2002 Salem Unit 1

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2, 5

Comments:

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
WS # 90	Group #	1	
	K/A #	076A2.01	
OK	Importance Rating		3.7

Ability to predict the impacts of loss of service water on the SWS and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions

Proposed Question: **"SRO ONLY" 90**

Given the following plant conditions:

- A loss of ALL normal Service Water Pumps has occurred due large amount of debris on the screens
- No Circ Water Pumps are available due to the debris

Which of the following describes the required operator action for the above condition?

A. Trip the Reactor and initiate E-0, Reactor Trip or Safety injection

B. Trip the Reactor, shut the MSIVs and initiate E-0, Reactor Trip or Safety injection

C. Trip the Reactor initiate manual Safety Injection and initiate E-0, Reactor Trip or Safety injection

D. Commence a rapid plant shutdown as long as temperatures remain below the trip setpoint

Proposed Answer:

B. Trip the Reactor, shut the MSIVs and initiate E-0, Reactor Trip or Safety injection

Explanation (Optional):

Technical Reference(s): AOP-SW-1 step 4.50 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:
- Bank # ____________
- Modified Bank # ____________ (Note changes or attach parent)
- New X

Question History:

Question Cognitive Level:
- Memory or Fundamental Knowledge ____________
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 5
- 55.43 5

Comments:

W590

4. SUBSEQUENT ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.1 __ Is essential header affected?	__ **GO TO** Step 4.4.
4.2 __ Are Backup SW Pumps available?	__ **GO TO** Step 4.4.
4.3 __ Start Backup SW Pumps as necessary to maintain 75 to 110 psig.	
4.4 **IAAT** SW Header pressure can **NOT** be maintained > 50 psig, **THEN PERFORM** <u>one</u> of the following based on system's ability to provide cooling: __ Trip the reactor <u>and</u> **INITIATE** E-0. __ **INITIATE** plant shutdown to MODE 3 using <u>one</u> of the following as applicable: • POP-2.1 (Operation At Greater Than 45% Power) • POP-3.1 (Plant Shutdown from 45% Power)	

NOTE

According to accident analysis, only a loss of the intake structure would cause a loss of all normal SW Pumps.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.5 __ **IAAT** a loss of all normal SW Pumps has occurred, **THEN GO TO** Step 4.50.	
4.6 __ **INITIATE** determination of affected header from pressure indications and individual component temperatures.	

3.3 IF alarm is from an instrument numbered from 1 through 15 (Turbine-Generator), THEN:

3.3.1 DISPATCH OPERATOR to check TCV-1102 (service water control valve on outlet of main oil coolers) for proper operation.

3.3.2 IF TCV-1102 is NOT functioning properly, THEN DIRECT OPERATOR to throttle open SWT-6, TCV-1102 Bypass Isolation, to return temperature to normal.

3.3.3 IF bearing oil drain temperature reaches 180°F, THEN:

3.3.3.1 IF P-8 is NOT illuminated, THEN:

3.3.3.1.1 TRIP the Reactor.

3.3.3.1.2 GO TO E-0, Reactor Trip or Safety Injection.

3.3.3.2 IF P-8 is illuminated, THEN:

3.3.3.2.1 TRIP the Turbine.

3.3.3.2.2 GO TO 3AOP-TURB-1, Main Turbine Trip Without a Reactor Trip.

3.3.4 IF bearing metal temperature on TIR-1101 or TIR-1102 reaches 225°F, THEN:

3.3.4.1 IF P-8 is NOT illuminated, THEN:

3.3.4.1.1 TRIP the Reactor.

3.3.4.1.2 GO TO E-0, Reactor Trip or Safety Injection.

3.3.4.2 IF P-8 is illuminated, THEN:

3.3.4.2.1 TRIP the Turbine.

3.3.4.2.2 GO TO 3AOP-TURB-1, Main Turbine Trip Without a Reactor Trip.

(CONTINUED ON THE NEXT PAGE)

NOTE

WHEN placing MBFP in service on condensate recirculation flow (wind milling), THEN bearing monitor alarm for thrust bearing temperature may occur due to uneven loading.

3.4 IF alarm is from an instrument numbered from 16 through 34 (Boiler Feed Pump No. 31 and 32), THEN:

3.4.1 DIRECT OPERATOR to throttle open SWT-16-1, (SWT-16-2), 31(32) MBFP Cooler Inlet Isolation, to return temperature to normal.

3.4.2 IF temperature continues to increase, THEN:

3.4.2.1 REDUCE main turbine load to less than 700 MWe per POP-2.1, Operation at Greater Than 45% Power.

3.4.2.2 REMOVE affected MBFP from service per SOP-FW-001, Main Feedwater System Operation.

3.4.3 IF either of the following occurs: THEN

- Oil temperature from the MBFP bearings increases to 190°F
- Turbine thrust pad metal temperature increases to 220°F

3.4.3.1 TRIP the affected MBFP.

3.4.3.2 GO TO 3AOP-FW-1, Loss of Feedwater.

3.4.4 IF both MBFPs are tripped, THEN GO TO E-0, Reactor Trip or Safety Injection.

3.5 IF alarm is from an instrument numbered from 35 through 52 (Circulating Water Pump No. 31 through 36), THEN:

3.5.1 IF motor bearing temperature increases to 198°F, THEN REMOVE affected circulating water pump from service per SOP-RW-001, Circulating Water System Operation.

(CONTINUED ON THE NEXT PAGE)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 91	Group #		2
	K/A #	034K4.02	
OK	Importance Rating		3.3

Knowledge of design feature(s) and / or interlock(s) which provide for fuel movement

Proposed Question: **"SRO ONLY" 91**

Which of the following describes what occurs when the Manipulator Crane INTERLOCK OVERRIDE Keyswitch is engaged?

A. Hoist Load Interlocks are bypassed except overload.

B. Gripper Interlocks are NOT bypassed to prevent dropping a fuel assembly.

C. Directly connects bridge, trolley and hoist controls to joystick; speeds are limited to 10 fpm.

D. Boundary Zone Interlocks are bypassed and Bridge/Trolley speed is limited to 30 fpm.

Proposed Answer:

C. Directly connects bridge, trolley and hoist controls to joystick; speeds are limited to 10 fpm.

Explanation (Optional):

Technical Reference(s): SD-17, page 34 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-FHD001 C (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43

Comments:

- Press for database info - accesses the database screen. The database contains the bridge and trolley coordinates for requested locations.
- Press for utilities - accesses the utility screen. Displays encoder selection information and allows changing the active encoder.
- Press for operation - returns the operator to the main menu.
- Logoff - allows the operator to log off from the system.
- Program manager - provides access to the Windows program manager. Requires SRO or system administrator access level.

2.2.5.8 Bypasses and Keyswitches

Depressing the BOUNDARY ZONE BYPASS pushbutton allows movement outside the normal boundary zone. Depressing the pushbutton for ten seconds seals in the interlock until the manipulator crane is returned to the boundary zone. Bridge and trolley speed is limited to a maximum of ten fpm. The pushbutton is backlighted yellow and "BOUNDARY LIMIT INTERLOCK" is visible and blinking when this bypass is active.

Depressing the HOIST LOAD BYPASS pushbutton allows operation of the hoist with an underload condition. The pushbutton is backlighted red and "LOAD BYPASS OVERRIDE ACTIVE" is visible and blinking when this bypass is active.

The INTERLOCK OVERRIDE keyswitch allows manipulator crane operation when it is necessary to place a fuel assembly or the crane in a safe condition. It bypasses all PLC control functions and directly couples the bridge, trolley, and hoist to their joysticks. The maximum speed is ten feet per minute. Gripper operation is allowed at any time, but the mechanical interlock prevents a loaded gripper from unlatching. "PLC OVERRIDE DETECTED" is visible and blinking when this interlock is active.

2.2.6 RCCA Change Fixture (Figure 17.0-17)

The RCCA change fixture is designed to remove rod cluster control and spider mounted secondary source assemblies from spent fuel assemblies and insert them into new or partially spent fuel assemblies. The RCCA change fixture is not normally used during a full core off-load. All insert moves are performed in the SFP.

The fixture consists of two main components: (1) a guide tube, permanently mounted to the reactor cavity wall, for containing and

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 92	Group #		2
	K/A #	035A2.03	
OK	Importance Rating		3.6

Ability to predict the impacts of pressure/level transmitter failure on the S/G and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations

Proposed Question: **"SRO ONLY" 92**

Given the following:

- Unit 3 operating at 100% power
- "B" channels of steam flow and feed flow in control
- Main Feed Regulating Valves in AUTO
- Steam Generator Pressure Channel PT-419B fails HIGH

Which of the below statements describes the plant response and the required actions to stabilize the plant?

A. 31 S/G controlling steam flow indication would increase, 31 S/G actual feed flow and level would increase, 31S/G level error would return level to program level but the operator should swap Steam Flow and Feed Flow for 31 S/G to "A" channel.

B. 31 S/G controlling steam flow indication would increase, 31 S/G actual feed flow and level would increase, the operator must swap Steam Flow for 31 S/G to "A" channel to prevent a Turbine Trip.

C. 31 S/G controlling steam flow indication would decrease, 31 S/G actual feed flow and level would decrease, 31S/G level error would return level to program level but the operator should swap Steam Flow and Feed Flow for 31 S/G to "A" channel.

D. 31 S/G controlling steam flow indication would decrease, 31 S/G actual feed flow and level would decrease, the operator must swap Steam Flow for 31 S/G to "A" channel to prevent a Turbine Trip.

Proposed Answer:

A. 31 S/G controlling steam flow indication would increase, 31 S/G actual feed flow and level would increase, 31S/G level error would return level to program level but the operator should swap Steam Flow and Feed Flow for 31 S/G to "A" channel.

Explanation (Optional):

Pressure compensation for steam flow will cause indicated flow to increase. The SF/FF mismatch is small enough for level error to compensate and return S/G level back to program thus prevent and Turbine Trip on High S/G level.

Technical Reference(s): 3-AOP-INSTR-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-ICSGL 5.0 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 5

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Unit Status A SG pressure channel failure has occurred.	
4.41 __ Has Channel C failed?	__ **GO TO** Step 4.43.

NOTE

Affected atmospheric steam dump must remain in manual control until Channel C instrument is restored to service.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.42 __ Place affected atmospheric steam dump in manual and return to appropriate position for plant conditions.	
4.43 __ Are both SG Transfer Switches selected to the non-affected channel?	__ Select both SG Transfer Switches to the non-affected channel.
4.44 Refer to the following TS tables for required actions: __ 3.3.1-1 __ 3.3.2-1	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
WS # 93	Group #		2
	K/A #	068A2.04	
OK	Importance Rating		3.3

Ability to predict the impacts of failure of automatic isolation on the Liquid Radwaste System and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations

Proposed Question: **"SRO ONLY" 93**

A liquid release of 32 Monitor Tank is in progress. A release permit was generated for the release and was approved. The following annunicators/conditions are received in the control room:

- R18, LIQUID EFF
- CHANNEL FAILURE
- R-18, Liquid Waste Effluent monitor is alarming
- The discharge remains in progress

Which one of the following describes the effect on the plant and the actions required?

A. The release should have automatically terminated. Stop the release and direct chemistry to sample the 32 Monitor Tank then re-calculate allowable release rate to determine if release may continue.

B. The release should have automatically terminated. Stop the release and re-verify the release permit calculations, release may resume provided calculations were correct.

C. R-18 monitor has failed. Request HP recheck calculations for liquid release and recommend corrective action that will be required per the ODCM.

D. R-18 monitor has failed. The release may continue provided two independent samples are taken to validate the release permit.

Proposed Answer:

A. The release should have automatically terminated. Stop the release and direct chemistry to sample the 32 Monitor Tank then re-calculate allowable release rate to determine if release may continue.

Explanation (Optional):

Technical Reference(s): ONOP-RM-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: I3LP-ILO-RMSPRM E (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

ATTACHMENT 1
RADIATION MONITOR
AUTOMATIC ACTIONS

Page 2 of 2

Monitor Number	Description	Automatic Action
R-15	Condenser Air Ejector Monitor	The air ejector discharge is directed to Containment 1. The "SJAE DIVERSION VALVE" diverts to Containment 2. The "SJAE BLOWER" starts. 3. CA-PCV-1229 & 1230 "SJAE EFFLUENT ISOLATION" OPEN 4. MS-PCV-1133 "Main Steam to hoggers Pressure Regulator" closes 5. MS-MOV-19 "Main Steam to Aux. Steam and Reboiler Isolation Valve" closes 6. 5EX-MOV-26 "Extraction Steam to Reboiler Isolation" closes
R-18	Liquid Waste Disposal Monitor	RCV-018 Liquid Waste Release Radiation Valve closes
R-19	Steam Generator Blowdown Monitor	1. S/G Blowdown Sample Valves close (this isolates the Rad Monitor) 2. S/G Blowdown Valves close 3. S/G blowdown flash tank city water supply closes
R-27	Wide Range Plant Vent Gas Monitor	1. WD-RCV-014 Waste Gas Discharge Valve closes 2. Containment Ventilation Isolation will be initiated 3. PAB exhaust will be diverted through the charcoal filters
R-56 A/B/C	Sewage Effluent Monitor	<u>NOTE</u> ST-7 can be operated by a switch on the north wall of the sewage holding tank building 1. <u>IF</u> R 56A, B, or C alarms, <u>THEN</u> ST-7 diverter valve will be activated and sewage will be pumped to the 35,000 gallon sewage holding tank <u>IF</u> R 56C alarms, <u>THEN</u> the upper lift station will shutdown
R-61	CPF Regen Release Monitor	1. WDL-AOV-32 and WDL-AOV-37 HTDS and LTDS discharge valves will close 2. WDL-AOV-31 and WDL-AOV-39 Recirculation Valves will open

-END OF ATTACHMENT-

ATTACHMENT 6
HIGH RADIATION ALARM ON R-18

1. IF **R-18** has ALARMED, THEN PERFORM the following:

   a. Place the permissive switch located on the Radiation Monitoring Control Panel in the CCR to the "Blocked" position to prevent inadvertent opening of RCV-018.

   b. Request the Watch Chemist sample the Monitor Tank that was being released and analyze for radioactivity.

   c. WHEN the Chemist sample results are obtained, THEN re-calculate allowable release rate and determine whether to release the tank or reprocess the tank through the Waste Disposal Facility.

   d. To release the Monitor Tank perform the following:

      1) Depress the "reset" pushbutton for R-18.

      2) Place permissive switch for R-18 to the "Unblocked" position.

      3) Have the Nuclear NPO open RCV-018.

-END OF ATTACHMENT-

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 94	Group #		1
	K/A #	2.1.11	
OK	Importance Rating		3.8

Knowledge of less than one hour technical specifications for a system

Proposed Question: **"SRO ONLY" 94**

A plant heatup/startup is in progress with RCS average temperature at 280°F. The following plant conditions develop:

- 31 and 32 RHR pumps become inoperable
- 31 and 32 SI pumps become inoperable

Which one of the following describes the Technical Specification Actions?

A. Restore only the SI pumps to OPERABLE status before reaching 350°F.

B. Restore the RHR pumps and only one SI pump to OPERABLE status before reaching 350°F.

C. Immediately initiate action to restore one of the RHR pumps to OPERABLE status.

B. Immediately initiate action to restore both SI pumps to OPERABLE status.

Proposed Answer:

C. Immediately initiate action to restore one of the RHR pumps to OPERABLE status.

Explanation (Optional):

Technical Reference(s): TS 3.5.3 Condition A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 2, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 95	Group #		1
	K/A #	2.1.33	
	Importance Rating		4.0

Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications

Proposed Question: **"SRO ONLY" 95**

Given the following:

- The plant is at 75% power.
- 31 CCW heat Exchanger is isolated and the appropriate Condition of TS 3.7.8 is entered
- Engineering reports that the CCW flow through 32 CCW Heat Exchanger is inadequate to supply appropriate cooling to both RHR Heat Exchangers.
- As such, the RHR trains are inoperable.
- Operations has isolated CCW to 31 RHR Heat Exchanger and documented proper flow capability to 32 RHR Heat Exchanger.

Per the attached Technical Specifications, do Technical Specifications require entry into Condition A of TS 3.5.2, including why?

A. Yes. LCO 3.0.1 requires all LCOs to be met including the 72-hour LCO of TS 3.5.2.

B. Yes. LCO 3.0.2 requires entry into TS 3.5.2 ACTIONS since LCO 3.5.2 is NOT met.

C. No. LCO 3.0.5 waives the requirement to enter TS 3.5.2 ACTIONS during OPERABILITY determinations.

D. No. LCO 3.0.6 waives the requirement to enter TS 3.5.2 ACTIONS provided the safety function is maintained.

Proposed Answer:

D. No. LCO 3.0.6 waives the requirement to enter TS 3.5.2 ACTIONS provided the safety function is maintained.

Explanation (Optional):

Technical Reference(s): TS 3.0.6
TS 3.5.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.5.2

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: 5/10/2004 Davis-Besse 1

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2, 3

Comments:

3.0 LCO APPLICABILITY (continued)

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Test Exception LCOs, such as 3.1.8, allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

LCO 3.5.2 Three ECCS trains shall be OPERABLE.

-----------------------------NOTES-----------------------------

1. In MODE 3, both HHSI flow paths may be isolated by closing the
...p to 2 hours to perform pressure
...g per SR 3.4.14.1.

EXAM
Attachment

2. ...th HHSI pumps made incapable of injecting
..., "Low Temperature Overpressure
... allowed for up to 4 hours or until the
... cold legs exceeds 375°F, whichever comes
...

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND Two HHSI pumps, one RHR pump and one Containment Recirculation pump are OPERABLE.	A.1 Restore train(s) to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed.	12 hours

Number	Position	Function
SI-856B	Closed	HHSI Loop 33 Hot Leg Injection Stop Valve
SI-856G	Closed	HHSI Loop 31 Hot Leg Injection Stop Valve
SI-1810	Open	RWST outlet isolation
AC-744	Open	Common discharge isolation for RHR pumps
SI-882	Open	Common RWST suction isolation for RHR pumps
SI-842	Open	HHSI pump minimum flow line isolation
SI-843	Open	HHSI pump minimum flow line isolation
SI-883	Closed	RHR pump return to RWST isolation
AC-1870	Open	RHR pump minimum flow line isolation
AC-743	Open	RHR pump minimum flow line isolation

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	Verify that each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.6	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Valve Numbers SI-856B SI-856G SI-2165 SI-2170 SI-856C SI-856H SI-2166 SI-2171 SI-856D SI-856J SI-2168 SI-2172 SI-856E SI-856K SI-2169	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.7	Verify, by visual inspection, each ECCS train containment sump suction inlet and recirculation sump suction inlet is not restricted by debris and the suction inlet screens show no evidence of structural distress or abnormal corrosion.	24 months

		RO	SRO
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 96	Group #		2
	K/A #	2.2.26	
OK	Importance Rating		3.7

Knowledge of refueling administrative requirements

Proposed Question: **"SRO ONLY" 96**

Which of the below is the first evolution that the Refueling Senior Reactor Operator must be stationed in the Vapor Containment Building?

A. Reactor Head Detensioning

B. Reactor Head Lift

C. Upper Internals Lift

D. Only during Fuel movement

Proposed Answer:
B. Reactor Head Lift

Explanation (Optional):

Technical Reference(s): RP-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______

55.43 5

Comments:

1.4.1.2. Refueling Senior Reactor Operator (RSRO)

A. RSRO is in charge of all refueling activities and SHALL directly supervise all core alterations in accordance with Technical Specifications. RSRO has complete control and authorization over all refueling activities and all personnel involved with any aspect of refueling process. RSRO has complete authorization to stop any activity affecting refueling process.

B. RSRO SHALL NOT have any additional duties that conflict with his refueling responsibilities.

C. RSRO is responsible for all refueling activities occurring on his shift. As SRO in charge of refueling activities, it is RSRO's responsibility to ensure all Technical Specification requirements and procedural prerequisites are satisfied.

D. RSRO SHALL ensure that refueling process is performed in accordance with approved refueling procedures.

E. RSRO SHALL ensure implementation of any emergency procedure required as a result of refueling operation and SHALL communicate condition to Control Room Supervisor (CRS).

F. RSRO SHALL be present and in a location that allows direct observation of following refueling activities:

- Any movement of RV Head
- Any movement of Upper Internals
- Control Rod latching, unlatching and testing
- Any movement of fuel into or out of RV

G. During fuel movement into or out of the RV, RSRO will normally position himself in Containment and position an RO, Reactor Engineer, or NPO in the Fuel Storage Building. IF RSRO must leave Containment, THEN fuel movement SHALL stop.

H. RSRO SHALL be only approval authority for use of any manipulator crane interlock bypass not specifically allowed by refueling procedure.

I. RSRO is responsible for ensuring that Shift Manager (SM) is cognizant of all on going refueling activities. In addition RSRO SHALL notify and obtain concurrence from SM with regard to all associated Technical Specification requirements.

J. As necessary, RSRO SHALL coordinate activities with SM and/or Control Room Supervisor (CRS), with regard to balance of plant activities which support or affect refueling operations.

K. RSRO SHALL be cognizant of any refueling anomalies or events and SHALL notify SM of any potentially reportable events associated with refueling operations.

L. RSRO has responsibility to notify SM and/or CRS of any significant change in plant conditions as a result of refueling process or of any unexpected problems or anomalies of a significant nature.

M. RSRO SHALL maintain a refueling logbook as per OD-5, Narrative Log Keeping.

N. RSRO SHALL verify that each fuel assembly inserted or withdrawn is at the correct core location.

1.4.1.3. <u>Refueling Maintenance Supervisors</u> (RMS)

A. RMS is in charge of reactor disassembly and reassembly including reactor auxiliary equipment as described in procedures RP-1, and RP-2 . RMS has complete authorization to stop any activity affecting the described processes.

B. RMS SHALL NOT have any additional duties that conflict with his disassembly/reassembly responsibilities.

C. RMS SHALL ensure that reactor and auxiliary equipment disassembly/reassembly is performed in accordance with approved refueling procedures.

D. RMS is responsible for ensuring the RSRO, CRS or SM is cognizant of all ongoing reactor and auxiliary equipment disassembly and reassembly.

E. RMS is responsible for notifying the SM or CRS of any significant change in plant conditions as a result of activities under his supervision or of any unexpected problems or anomalies of a significant nature.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 97	Group #		2
	K/A #	2.2.24	
	Importance Rating		3.8

Ability to analyze the affects of maintenance activities on LCO status

Proposed Question: **"SRO ONLY" 97**

The following plant conditions exist:

- 100 % power
- 31 EDG is out of service for preventative maintenance

The maintenance supervisor requests a work permit for 33 Safety Injection (SI) pump.

Should 33 SI pump be taken out of service? Select the proper action to be taken with the justification for your choice.

A. Yes. Technical Specifications allow for one SI pump to be out of service, provided it is returned to service within 72 hours and the remaining 2 SI pumps are demonstrated operable.

B. Yes. There are no restrictions applicable to the current plant conditions concerning the removing from service 31 SI pump.

C. No. Technical Specifications require 3 SI pumps together with their associated piping and valves to be operable for all conditions except low power physics testing.

D. No. Technical Specifications state that if one EDG is out of service then the other 2 EDG's and their associated safeguards equipment must be operable within 4 hrs.

Proposed Answer:

D. No. Technical Specifications state that if one EDG is out of service then the other 2 EDG's and their associated safeguards equipment must be operable within 4 hrs.

Explanation (Optional):

Technical Reference(s): TS 3.8.1, Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ______
55.43 ______

Comments:

ACTIONS (continued)

B.2

safeguards power train (and DG). However, if a required safety feature is supported by an inoperable DG and the redundant safety feature that is powered from a different safeguards power train is also inoperable, then a loss of offsite power will result in the loss of a safety function. Required Action B.2 ensures that appropriate compensatory measures are taken for a Condition where the loss of offsite power could result in the loss of a safety function when a DG is not OPERABLE.

The turbine driven auxiliary feedwater pump is not required to be considered a redundant required feature, and, therefore, not required to be determined OPERABLE by this Required Action, because the design is such that the remaining OPERABLE motor driven auxiliary feedwater pumps is capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature powered from another safeguards power train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with either OPERABLE DG, results in starting the Completion Time for the Required Action. A COMPLETION TIME of four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 98	Group #		3
	K/A #	2.3.6	
OK	Importance Rating		3.1

Knowledge of the requirements for reviewing and approving release permits

Proposed Question: **"SRO ONLY" 98**

Which ONE of the following can provide final authorization for a Liquid Rad Waste release?

A. Only the Shift Manager

B. Only the Shift Manager or Control Room Supervisor

C. Only the Shift Manager or Chemistry Supervisor

D. Only the Shift Manager or HP Supervisor

Proposed Answer:
B. Only the Shift Manager or Control Room Supervisor

Explanation (Optional):

Technical Reference(s): 3-SOP-WDS-014, Attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ________________ (As available)

Question Source: Bank # INPO 19844

Modified Bank # ________________ (Note changes or attach parent)

New ________________

Question History: 10/29/2001 Braidwood 1

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________________

10 CFR Part 55 Content: 55.41 ________

55.43 4

Comments:

ws 78

ATTACHMENT 1
LIQUID RADIOACTIVE WASTE RELEASE PERMIT FORM
(Page 1 of 1)

Release ID:		Volume: gal. Or Flow rate	Permit #:	
Recirc. Started - Date: Time:		Min. Recirc. Time (**T**): ______ hr.	Earliest Sample: Date: Time:	
Radiation Monitor #:	In Service (Circle one): **YES** **NO**	Rad Monitor Source Check: **SAT** **UNSAT**	Available dilution flow for release (**B**): gpm	
Chemistry analysis Sample #: Date: Time: Boron ppm: Boron pounds:			Total Gamma Activity (**C**): µCi/ml	
Allowable Diluted Concentration (**ADC**) in discharge canal: µCi/ml			Permissible Chemistry Discharge Rate (**Dc**): gpm	
Permissible Radioactive Discharge Rate (**Dr**): gpm		Most Restrictive Discharge Rate (**D**): gpm	From: (Check one)	☐ Waste Chemistry ☐ Radioactivity
Radiation Monitor:	Alert Setpoint: µCi/ml	Calculated Alarm Setpoint (**R**): µCi/ml	Actual Alarm Setpoint: µCi/ml	
IF Rad Monitor is Out Of Service	Monitor taken out of service on: (Max. 30 days) Date: Time:		2nd sample obtained and analyzed by:	
	Release calculations performed by:		Release calculations verified by:	
Release Authorization	(1) Release valve alignment reviewed (if applicable) (2) Release calculations reviewed (3) Release authorized by: (SM/CRS)			
	Release initiated: Date: Time:		Rad Monitor reading during release: µCi/ml	
Discharge flow meter operable: (Circle one) **YES** **NO***	*IF flow meter is OOS, THEN RECORD estimate approx 1hr into release and then every 4 hours (required by IP-SMM-CY-001, Radioactive Effluents Control Program): **1 hr** **4 hr** **8 hr**			
Post-Release Section	Release terminated: Date: Time:		Actual volume released: gal	
Alert and Alarm setpoints reset per 3-SOP-RM-010: **YES** **N/A**			Performed By: (initials)	
COMMENTS:				

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 99	Group #		4
	K/A #	2.4.41	
OK	Importance Rating		4.1

Knowledge of emergency action level thresholds and classifications

Proposed Question: **"SRO ONLY" 99**

A Steam Generator Tube Rupture occurred on 34 Steam Generator. Prior to RCS depressurization and SI termination, 34 SG Atmospheric Steam Dump valve was periodically lifting due to high SG pressure. What Emergency Action Level declaration should be made for this event?

A. NUE

B. Alert

C. SAE

D. GE

Proposed Answer:

B. Alert

Explanation (Optional):

Technical Reference(s): IP-EP-120 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # ______________

Modified Bank # ______________ (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge ______________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 5

Comments:

9.1 - Emergency Action Levels

CATEGORY 3.0 REACTOR COOLANT SYSTEM

Category	General	Site Area	Alert	Unusual Event
3.1 RCS Leakage		**3.1.3 {>200°F, ≤ 200°F }** RVLIS cannot be maintained [Unit 2] **> 41%** [Unit 3] **> 33%** with no RCPs running OR With the reactor vessel head removed, it is reported that water level in the Reactor Vessel is dropping in an uncontrolled manner and core uncovery is likely	**3.1.2 {>200°F}** Primary system leakage exceeding capacity (**> 75 gpm**) of single charging pump	**3.1.1 {>200°F}** Unidentified or pressure boundary leakage **> 10 gpm** OR Identified leakage **> 25 gpm**
3.2 Primary to Secondary Leakage		**3.2.2 {>200°F}** Unisolable release of secondary side to atmosphere from the affected steam generator(s) with primary to secondary leakage exceeding capacity (**> 75 gpm**) of a single charging pump **3.2.3 {>200°F}** Unisolable release of secondary side to atmosphere from the affected steam generator(s) with primary to secondary leakage > Technical Specification limit in any steam generator AND Coolant activity **> 300 µCi/cc** of I-131 equivalent		**3.2.1 {>200°F}** Unisolable release of secondary side to atmosphere from the affected steam generator(s) with primary to secondary leakage > Technical Specifications limit in any steam generator
3.3 RCS Subcooling			**3.3.1 {>200°F}** RCS subcooling < SI initiation setpoint due to RCS leakage	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		G
WS # 100	Group #		4
	K/A #	2.4.28	
OK	Importance Rating		4.3

Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: **"SRO ONLY" 100**

Assume the operators have just started a depressurization of the intact S/G's per Step 11 of FR-C.1, Response To Inadequate Core Cooling with the following indications:

- Core exit TCs at 1250°F and decreasing
- SG pressures 900 psig and decreasing
- RWST level just decreased to 11.5 ft.

Select the appropriate action for the above conditions.

A. Continue in FR-C.1 until directed to E-1, Loss of Reactor or Secondary Coolant, then Transfer to ES-1.3, Transfer to Cold Leg Recirculation.

B. Complete step 11 i.e.; SG<125 psig and Th less than 350°F, then transfer to ES-1.3, Transfer to Cold Leg Recirculation. Initiate cold leg recirculation, then return to FR-C.1.

C. Immediately transfer to ES-1.3, Transfer to Cold Leg Recirculation while continuing SG depressurization. Initiate cold leg recirculation, then return to FR-C.1 step 11.

D. Transfer to ES-1.3, Transfer to Cold Leg Recirculation as soon as core exit TC less than 1200°F. Initiate cold leg recirculation, then return to FR-C.1 step 11.

Proposed Answer:

C. Immediately transfer to ES-1.3, Transfer to Cold Leg Recirculation while continuing SG depressurization. Initiate cold leg recirculation, then return to FR-C.1 step 11.

Explanation (Optional):

Technical Reference(s): ______ (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # INPO
Modified Bank # 27336 (Note changes or attach parent)
New

Question History: 4/27/2004 Ginna 1

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 2

Comments:

4.3.17 Certain contingency EOPs take priority over FRPs due to specific initiating events. These procedures are identified by a note at the beginning of the EOP:

- ECA-0.0 LOSS OF ALL AC POWER
- ECA-0.1 LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED
- ECA-0.2 LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED
- ES-1.3 TRANSFER TO COLD LEG RECIRCULATION

4.3.18 In general, IF an ORANGE or RED condition develops and subsequently clears during execution of a procedure, THEN it is not necessary to enter the FRP for the ORANGE or RED condition at conclusion of current procedure. The exception is the INTEGRITY FRP, which should be entered to establish appropriate soak time, UNLESS it can be determined that the alarm cleared within 25 minutes of initiation.

4.3.19 IF the CFMS is available for monitoring, THEN it is acceptable to use the CFMS to monitor the status trees. **(IP3)**

4.4 CONTROL ROOM USAGE OF THE EOP NETWORK

4.4.1 Entry into the EOPs is limited to the following conditions:

- WHEN the reactor is in Hot Standby or greater AND any Reactor Trip or Safety Injection occurs OR is required, THEN E-0, REACTOR TRIP OR SAFETY INJECTION, SHALL be entered, unless the Control Room has been evacuated OR a complete loss of all AC Safeguards buses has occurred.

- WHEN the reactor is in Hot Shutdown or greater AND a complete loss of power on all AC Safeguards buses occurs, THEN ECA-0.0, LOSS OF ALL AC POWER, SHALL be entered, unless the Control Room has been evacuated. This entry condition also applies during performance of ANY other EOP.

- IF the Control Room has been evacuated, THEN AOP-SSD-1, CONTROL ROOM INACCESSIBILITY SAFE SHUTDOWN CONTROL, SHALL take priority over all EOPs.

- ATTACHMENT 1, EOP APPLICABILTY, presents the overall mode applicabilty for each EOP.

CAUTION: If RWST level decreases to less than (U.02), the SI System should be aligned for cold leg recirculation using ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

PURPOSE: To guarantee coolant flow to the core by switching to cold leg recirculation if the RWST level decreases below the switchover setpoint

BASIS:

If the switchover level in the RWST is reached, which could happen at any time during the course of guideline FR-C.1 depending upon the amount of RCS inventory losses, the operator should immediately go to ES-1.3, TRANSFER TO COLD LEG RECIRCULATION to maintain coolant flow to the core. When RWST level decreases to (U.02), there should be sufficient water available in the recirculation sump to switch the suction supply to the SI pumps. The remainder of RWST water is reserved for spray pump usage.

ACTIONS:

Determine if RWST level decreases to less than (U.02)

INSTRUMENTATION:

RWST level indication

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

(U.02) RWST switchover setpoint in plant specific units.