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Question Number:	SRO 01
Question:	The plant is at 99 % power. All control systems are operating in AUTO, with the exception of Rod Control, which is operating in MANUAL. Control Bank D is at 198 steps. The Primary Board Operator withdraws Control Bank D two steps to maintain Tave on program. When the In/Hold/Out switch is released, Control Bank D rods continue to move in the OUT direction. What action should be taken? IAW procedure A. Trip the reactor, go to E-0, REACTOR TRIP OR SAFETY INJECTION. B. Attempt to reinsert Control Bank D to 198 steps. If Control Bank D will not stop moving, trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. C. Check Tave/Tref indication. If Tave is > Tref by more than 1°F, adjust turbine load to restore Tave/Tref deviation to within 1°F. D. Check Tave/Tref indication. If Tave is > Tref by more than 1°F, borate to restore Tave/Tref deviation to within 1°F.
Answer:	A.
Justification:	During an inadvertent rod withdrawal, OS1210.04 directs the crew to place control rods in Manual. If they continue to move, the procedure directs the crew to trip the plant. Since control rods are in Manual as part of the initial conditions, if they continue to move after withdrawal is stopped, the plant should be tripped. Distractors C and D do not require a reactor trip, but rather a diagnosis of the event. Distractor B calls for action not covered by the procedure before tripping the plant. Distractor C calls for action taken during a dropped rod event
Direct/New/ Modified	Direct from 1998 Seabrook NRC exam
K/A #:	000001AA2.03
K/A Values:	4.5/4.8
Cognitive Level:	Analysis (III)
References:	OS1210.04, INADVERTENT ROD WITHDRAWAL Lesson plan L1184I, Objective L1184I04RO

What immediate action...

D

A or B

Implausible

Question Number:	SRO 02
Question:	The following conditions exist: • All control systems in AUTOMATIC An event occurs, causing a drop in reactor power to 97% followed by an increase to 99% • Tave DECREASED 2°F • Control Bank D is at 222 steps and withdrawing Which of the following events may have caused these indications? A. A Main Turbine Control valve inadvertently closed. B. A control rod has dropped. C. Inadvertent control rod withdrawal. D. A Main Steam safety valve is leaking.
Answer:	B.
Justification:	B is correct for the indications given. D is incorrect because power would not drop on a MSSV leak. A is incorrect because Tave would increase for a control valve closure. C is incorrect because the normal response to a drop in Tave would be for control bank D to withdraw.
Direct/New/ Modified	Direct from 1998 Seabrook NRC exam.
K/A #:	000003AA2.01
K/A Values:	3.9
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1410I, Objective L1410I05RO, section 3.3.B

~~Why not a backward logic?~~

~~A rod drops what is the effect on Power, Tave, Rod~~

Used on previous exam leave as is

Question Number:	SRO 03
Question:	During performance of OX1410.02, QUARTERLY ROD OPERABILITY CHECK AND MONTHLY NEW FULL OUT POSITION SURVEILLANCE, Rod F2 in Control Bank 'B', group 1 stops moving when it is 14 steps from it's new "Full Out" bank position. I&C reports that the Lift Coil has failed and the rod is declared INOPERABLE. Technical Specification 3.1.3.1, ACTION b.3.d limits reactor power to 75% Rated Thermal Power. Which of the following is the reason for this power limit? A. Acceptable power distribution is assured and continued operation is allowed if the rod is declared untrippable. B. Allows the plant to be operated without performing a re-evaluation of the safety analysis affected by a misaligned rod. C. Relieves the operators of having to calculate shutdown Margin every 12 hours. D. Provides assurance of fuel rod integrity during continued operations.
Answer:	D
Justification:	A is incorrect because a 6 hour shutdown is required per action a if the rod is declared untrippable. B is incorrect because the T.S. does require a reevaluation of the safety analysis affected by a misaligned rod. C is incorrect because the T.S. requires the Shutdown Margin be calculated every 12 hours until the rod is realigned. D is correct from the basis for T.S 3.1.3.1.
Direct/New/ Modified	Modified from 1996 SRO exam
K/A #:	000005K3.04
K/A Values:	4.1
Cognitive Level:	Analysis (III)
References:	Technical Specifications 3.1.3.1 and its associated basis.

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Question Number:	SRO 04
Question:	The following conditions exit: • No forced or natural RCS circulation flow • Inadequate core cooling exists following a LOCA • The crew is performing actions of FR-C.1, RESPONSE TO INADEQUATE CORE COOLING Which of the following is the primary reason for restoring Narrow Range level in at least one intact steam generator to greater than 5% (25% for adverse containment)? A. Ensures SG level is above the U-Tubes for adequate "Iodine Partitioning" B. Maintains intact SG(s) available as heat sink. C. Keeps the feed ring covered to prevent water hammer. D. Maintains the SG tubes covered to prevent thermal gradients from forming.
Answer:	B
Justification:	A is incorrect as this would be a consideration for a SGTR not an ICC condition. B is correct from the basis document for step #9 of FR-C.1. C is incorrect, while this may be a secondary or tertiary benefit of maintaining adequate NR water level it is <u>NOT</u> the primary purpose of this procedure step in FR-C.1. D is incorrect, this is a consideration for maintaining adequate NR level during a SGTR event not an ICC condition.
Direct/New/ Modified	New
K/A #:	000011EK1.01
K/A Values:	4.1/4.4
Cognitive Level:	Comprehension (II)
References:	FR-C.1 and background document

Primary

Different SG NR level criteria dependent upon accident

Question Number:	SRO 05
Question:	The following plant conditions exist: • A rupture in the piping downstream of SI-V138/139 has occurred • The check valves on the piping connecting to the RCS have failed causing a LOCA into the Containment penetration area of the PAB • The Reactor has tripped and Safety Injection has actuated on Low PZR Pressure • All ECCS systems are operating as designed. • The crew transitions to ECA-1.2, LOCA OUTSIDE CONTAINMENT, at step # 25 of E-0 based upon abnormal PAB radiation level. Assuming plant conditions do not significantly change and the leak is unisolable, what procedure in the EOP network will ultimately be used to deal with this accident? A. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION. B. ECA-1.2, LOCA OUTSIDE CONTAINMENT C. E-1, LOSS OF REACTOR OR SECONDARY COOLANT. D. ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.
Answer:	D
Justification:	A is incorrect as there are no procedural transition criteria to ES-1.2 in this condition. B is incorrect as ECA-1.2 contains actions that assume LOCA outside containment is in the RHR system. This is not the case and ECA-1.2 will direct transition to ECA-1.1. C is incorrect as question stem indicates that LOCA outside containment is unisolable and therefore a transition to E-1 would not be made. D is correct. The crew should transition to ECA-1.1 from ECA 1.2, Step 4 RNO due to inability to isolate leak. ECA-1.1 contains the long term actions required to cooldown and depressurize the RCS when emergency coolant recirculation is unavailable.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	E04EK1.3
K/A Values:	3.9
Cognitive Level:	Analysis (III)
References:	E-0, Step 25 ECA-1.2, entry conditions and step 4 ECA-1.1, entry condition #3 Objective L1209I03SR

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Question Number:	SRO 06
Question:	A small break LOCA has occurred. Automatic SI is actuated but the reactor does not trip. In accordance with FR-S.1, the crew shuts the reactor down using manual rod insertion and emergency boration. The emergency boration is continuing. The crew transitions to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, from E-0, Step #20. At Step #8 of E-1 the Primary Operator is directed to reset SI to enable stopping of the RHR pumps. SI will <u>NOT</u> reset. What is ~~a~~ the possible cause of the SI reset failure? A. The initiating condition causing the SI actuation has not cleared. B. The Reactor Trip Breakers are closed. C. The timer in the Safety Injection Block/Reset logic has not timed out. D. The automatic reactor trip signal has cleared.
Answer:	B.
Justification:	A is incorrect because SI may be reset as long as the 60 second timer has timed out and P-4 is energized, whether the initiating condition is still active or not. C is incorrect because the procedure flowpath to get to ES-1.1 would be much longer than 60 seconds. D is incorrect because at least one automatic trip signal will remain active in these conditions and in any case the Rx trip breakers never opened. B is correct as P-4 inputs to the SI reset circuitry.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	E02EK2.1
K/A Values:	3.4/3.9
Cognitive Level:	Analysis (III)
References:	IS system text, figure 3.4 RP system text, section 4.4.1.1, page 28 Lesson Plan L1138I, Objective L1138I21RO

~~Backward Logic~~ — ~~possible~~ Real life situation expected to solve

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Question Number:	SRO 07
Question:	The plant is at 30 % power during a Plant Startup with all control systems in AUTOMATIC. RCP 'A' trips on Phase Differential Overcurrent. Assuming no operator action, which of the following describes the response of the plant to the RCP trip? A. Steam flow decreases in all SGs. All SG levels initially decrease, then increase as the secondary plant stabilizes and SGWLC responds. Control Rods withdraw to maintain Tave on program. B. Steam flow decreases in all SGs. SG 'A' level initially increases due to overfeeding. SG 'B', 'C', 'D' levels initially decrease due to increased steam demand. SG levels return to normal as SGWLC responds. Tave remains unaffected because Reactor power remains unaffected. C. Steam pressure decreases in all SGs. SG 'A' level decreases due to the loss of heat input. SG 'B', 'C', 'D' levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Tave and Tref stabilize at a lower value. D. Steam pressure decreases in all SGs. SG 'A' level decreases due to the loss of heat input. SG 'B', 'C', 'D' levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Control rods withdraw to return Tave to it's previous value.
Answer:	C
Justification:	A is incorrect because Steam flow increases on the 'B', 'C', & 'D' SGs to make up for the loss of steam flow from 'A' SG. Also 'A' SG NR level will initially decrease from shrink affect. B is incorrect because the SG levels will respond in an opposite manner to that described because of shrink (A SG) and swell (B, C, D SGs), and steam flow decreases from the 'A' SG. C is correct. Tref will decrease as Pimp decreases with Tsat in the SGs. Rod control will function to restore Tave to the new lower programmed value. D is incorrect. Rod control will not restore Tave to it's previous value because Tref has decreased, changing the Tave program (it will be a lower temperature).
Direct/New/ Modified	Modified from 1998 NRC exam.
K/A #:	000015/000017AK1.04 2.1.7
K/A Values:	2.9/3.1 3.7/4.4
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1405I Abnormal Transient Analysis Objectives L1405I01RO and L1405I02RO

Question Number:	SRO 08
Question:	The plant is operating at 100 % power when a Loss of Off-Site power causes a reactor trip. Two minutes following the trip, the following conditions exist: • All 4 S/G pressures trending slowly upward toward ASDV lift setpoint. • Core Exit Thermocouple temperatures are slowly increasing. • RCS Cold leg temperatures are slowly increasing. • RCS Hot leg temperatures are slowly increasing. Based on the above indications, what is the condition of the RCS? A. Natural Circulation has developed. Heat removal is being maintained by the condenser steam dumps. B. Natural Circulation has not developed. Heat removal may be established by opening the condenser steam dumps. C. Natural Circulation has developed. Heat removal is being maintained by atmospheric steam dumps. D. Natural Circulation has not developed. Heat removal may be established by opening the atmospheric steam dump valves.
Answer:	D
Justification:	A and B are incorrect because on a loss of off site power, condenser steam dumps are unavailable due to loss of circulating water pumps. C is incorrect based on expected indications for natural circ flow in Appendix A of ES-0.1. D is correct because question stem conditions indicate that natural circ has not developed based on criteria in appendix A of ES-0.1. Dumping steam with ASDVs will help establish natural circ cooling.
Direct/New/ Modified	Modified from 1998 NRC Exam
K/A #:	E09EK2.2
K/A Values:	3.6/3.9
Cognitive Level:	Synthesis (III)
References:	ES-0.1, appendix A Lesson Plan L1200I, Objective L1200I08RO

Question Number:	SRO 09
Question:	The crew is performing a rapid power decrease at BOL from 100% power to 30% power. Control Rods are in AUTO. Load reduction is being performed using turbine LOAD LIMIT. The following alarms are received: • D7761 CTL ROD BANK D INSERTION LIMIT LOW • D7762 CTL ROD BANK D INSERTION LIMIT LO-LO The crew initiates rapid boration per OS1202.04. Assuming the turbine load decrease rate remains constant throughout the event, which of the following describes the effects of the boration on the plant? A. Reactor power will DECREASE. Control rods will insert at a FASTER rate. Tave will INCREASE B. Reactor power will INCREASE Control rods will insert at a SLOWER rate Tave will DECREASE C. Reactor power will DECREASE Control rods will insert at a SLOWER rate Tave will DECREASE D. Reactor power will DECREASE Control rods will insert at a FASTER rate Tave will INCREASE the same
Answer:	C.
Justification:	Normal load reduction, control rods will try to match Tave and Tref. With a boration, the control rods do not have to add as much negative reactivity to keep Tave and Tref matched as power decreases. A is incorrect because rods will insert at a slower rate because boration will decrease power mismatch between primary and secondary, and therefore, heat input. B is incorrect because reactor power will be decreasing throughout the event. D is incorrect because Tave will also decrease due to the boration decreasing the power mismatch
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000024AK1.02
K/A Values:	3.6/3.9
Cognitive Level:	Synthesis (III)
References:	OS1202.04, OS1000.06, figure 6. Lesson Plan L1183I, Objective L1183I02RO

Question Number:	SRO 10
Question:	The following plant conditions exist: • Train 'A' RHR is operating in the shutdown cooling mode. • RCS temperature is 320°F and stable • RCP-1C is operating • MPCS VAS B4787 PCCW HD TK LVL RATE OF CHANGE HIGH is in alarm. • TRN 'A' PCCW HEAD TANK level is decreasing WHICH of the following is the cause of the noted conditions? A. A tube leak in the CVCS regenerative heat exchanger. B. A tube leak in a Seal Water Return heat exchanger. C. A leak in the 'C' RCP thermal barrier heat exchanger. D. A tube leak in the 'A' RHR heat exchanger.
Answer:	B
Justification:	B is correct because the PCCW operating pressure is higher than the seal return pressure. A is incorrect because the PCCW is not supplied to the CVCS regenerative heat exchanger. C & D are incorrect because the RCS pressure required to support RCP operation is greater than the operating pressure of PCCW.
Direct/New/ Modified	Modified from bank
K/A #:	026AA2.01
K/A Values:	3.5
Cognitive Level:	Analysis (III)
References:	P&ID 1-CC-B2020A, CC Objective: L1118I03RO

~~Backward logic?~~ Diagnostic

~~Backward Logic?~~

Solve

OK

Question Number:	SRO 11
Question:	At step 4 of FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, the operator checks RCS pressure less than 2385 psig. If NOT, the operator is directed to open the PORVs until RCS pressure is less than 2185 psig. What is the basis for this action? A. Ensures RCS pressure does NOT rise to the safety valve lift setpoint. B. Ensures no RCS mass is being lost through unnecessary automatic PORV operation. C. At 2385 psig, charging flow into the RCS is assumed to be insufficient. D. At 2385 psig, for the design basis ATWS, RCS integrity is being challenged.
Answer:	C
Justification:	Background document assumes 2385 psig is pressure too high for adequate boration flow (C correct). PZR safety valve setpoint is at higher pressure than this but action to open PORV is NOT to prevent reaching this setpoint (A incorrect). B is incorrect, if RCS pressure is 2385 psig the PZR PORVs SHOULD be open. D is incorrect, the design basis pressure is well above 2385 psig, (approximately 3200 psig)
Direct/New/ Modified	New
K/A #:	000029EK3.12,
K/A Values:	4.4/4.7
Cognitive Level:	Comprehension (II)
References:	FR-S.1, background, step 4 Objective L1200I11RO

? Manory

Question Number:	SRO 12
Question:	Plant Conditions: • A plant event has resulted in implementation of ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS. • Bus E-6 is de-energized leaving the steam driven EFW pump as the only source of EFW flow to the steam generators. • NR level in all four S/Gs is off scale low. • The operators have throttled EFW flow to 25 gpm per S/G in accordance with step #2 of ECA-2.1 causing a RED path on heat sink. • The steam driven EFW pump subsequently trips. D required What action should the crew take? I4 A. Continue with ECA-2.1. The caution prior to step #1 prohibits transition to FR-H.1. B. Attempt to place the startup feedpump in service while continuing with ECA-2.1. C. Transition to FR-H.1. RESPONSE TO LOSS OF SECONDARY HEAT SINK. D. Continue with ECA-2.1 SG NR levels are adequate.
Answer:	C
Justification:	A is incorrect, the caution prior to ECA-2.1, step #1 states that FR-H.1 should only be implemented if EFW flow capability of 500 gpm is not available. This condition is met and transition to FR-H.1 should be made. B is incorrect, there are no contingency actions in ECA-2.1 to place SUFP in service. C is correct, RED path is valid since no EFW flow is available. D is incorrect, S/G NR levels are NOT adequate as indicated in the stem. Continuation with actions of ECA-2.1 is not appropriate with inadequate EFW flow.
Direct/New/ Modified	Modified from 1996 NRC Exam
K/A #:	E12EA2.2
K/A Values:	3.9
Cognitive Level:	Comprehension (II)
References:	ECA-2.1 FR-H.1 Objective L1207I04RO

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Question Number:	SRO 13
Question:	Which of the following is characteristic of an event that would require entry to FR-P.1, RESPONSE TO PRESSURIZED THERMAL SHOCK? A. RCS PRESSURE DECREASE followed by rapid RCS HEATUP B. Rapid RCS COOLDOWN followed by RCS PRESSURE DECREASE C. Rapid RCS COOLDOWN followed by rapid RCS HEATUP D. Rapid RCS COOLDOWN followed by RCS PRESSURE INCREASE
Answer:	D.
Justification:	A and B are incorrect because a pressure decrease will lower the probability of a PTS event. C is incorrect because a rapid RCS heatup does not necessarily mean the RCS will repressurize, and heatup will cause vessel metal temp to be further away from RTNDT. D is correct and contains 2 components required for a PTS. The other 2 components are embrittlement and pre-existing flaw, which cannot be monitored.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	E08EK3.1
K/A Values:	3.8/3.9
Cognitive Level:	Comprehension (II)
References:	Lesson plan L1142I, Objective L1142I01RO

ok

?, ~~Memory~~

Standard supports their perspect

Question Number:	SRO 14
Question:	PLANT CONDITIONS: • Reactor Power is 32 % • Turbine load has been reduced from 900 MWE to 360 MWE • Condenser Vacuum is 23.5 inches Hg and slowly decreasing Based on the above indications, which action is the crew required to take? IAW A. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. B. Continue the load decrease to increase condenser vacuum to > 25 inches Hg. C. Trip the turbine and go to ON1231.02, TURBINE TRIP BELOW P-9. D. Remove the turbine generator from service IAW OS1000.06, POWER DECREASE.
Answer:	A
Justification:	A is correct because a manual Rx trip is required when load has been decreased to 360 MWE and vacuum cannot be maintained >25" Hg. B is incorrect because load reduction should not continue below 360 MWE. C is incorrect. A reactor trip is required and power is > P-9 (20%) which will cause a turbine trip. D is incorrect because load was originally > 360 MWE.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	000051AA2.02
K/A Values:	3.9/4.1
Cognitive Level:	Synthesis (III)
References:	Objective L1180I08RO ON1233.01, LOSS OF CONDENSER VACUUM

Question Number:	SRO 15
Question:	A Loss of All AC power has occurred. The crew has entered ECA-0.0, LOSS OF ALL AC POWER. An operator has been dispatched to perform Attachment 'A' to shed DC loads. When Attachment 'A' is completed, which of the following loads will still be energized? A. ED-PP-12E, Non-Vital Instrument Distribution Panel 12E B. EDE-PP-1A, Vital Instrument Distribution Panel 1A C. Bus E61 Auxiliary Bus D. CP-CP-111, Reactor Trip Switchgear
Answer:	B.
Justification:	Vital 120V AC panels remain energized on load stripping. A, C, and D are all listed as being de-energized per attachment 'A'.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000055EA1.04
K/A Values:	3.5/3.9
Cognitive Level:	Memory (I)
References:	ECA-0.0, Attachment 'A' Lesson Plan L1201I, Objective L1201I13RO

Question Number:	SRO 16
Question:	During his routine rounds in the 'A' train essential switch gear room the secondary NSO notices that the "Reverse Transfer" lamp is lit on static transfer switch EDE-CP-1E. What is the condition of EDE-PP-1E? A. The transfer switch has swapped EDE-PP-1E back to its inverter supply. B. The transfer switch has swapped EDE-PP-1E to its alternate supply. C. The maintenance supply breaker at EDE-PP-1E has tripped open. D. EDE-PP-1E is de-energized.
Answer:	B
Justification:	A is incorrect because the reverse transfer lamp indicates that the transfer switch has swapped the power panel to its alternate power supply. B is correct, lamp indicates that EDE-PP-1E is on alternate (non-inverter) power. C is incorrect because the maintenance supply breaker to the panel is normally open. D is incorrect because the reverse transfer light lit indicates that alternate power supply power is being provided to the power panel and no other indicator/alarms which would implicate a loss of power to the panel are indicated in the question stem.
Direct/New/ Modified	Modified from 1996 NRC Exam
K/A #:	000057AA2.06
K/A Values:	3.7
Cognitive Level:	Memory (I)
References:	120 VAC vital instrumentation distribution System detailed system text. VAS alarm response for D5734 Objective L1098I09RO

Question Number:	SRO 17
Question:	The liquid radwaste test tank discharge radiation monitor (R-6509) has been declared INOPERABLE. Which of the following describes the Technical Specification ACTION ~~that will~~ permit continued release from the liquid waste system? A. Liquid waste discharge will not be permitted until the discharge radiation monitor is returned to OPERABLE status. B. A temporary monitor may be used provided its alarm setpoint is more conservative than the R-6509 setpoint to allow the operator sufficient time to manually secure the discharge in the event an alarm condition occurs. C. Two independent samples of the tank to be discharged must be analyzed, and two technically qualified staff members must independently verify the release rate calculations and the discharge line valve lineup. D. Samples must be taken every 15 minutes while the discharge is in progress, to verify the effluent is within Technical Specification limits.
Answer:	C.
Justification:	Answer A is incorrect because the discharge can continue provided the requirements listed in answer C are met. Answer B is incorrect because no credit is taken for the use of temporary monitors. The plant does have temporary monitors, but they would not be used as discussed in this distractor. Answer D is incorrect because the frequency of the sample is not in T.S.
Direct/New/ Modified	Direct from 1998 NRC exam.
K/A #:	000059AK3.01
K/A Values:	3.9
Cognitive Level:	Memory (I)
References:	Tech Spec 3.3.3.9 Lesson Plan L1141I, Radiation Data Mgmt program, Objective L1141I14RO

Delete

TS provided?

Question Number:	SRO 18
Question:	Plant conditions: • 'A' Train Service Water (SW) was transferred from the ocean to the cooling tower for quarterly surveillance testing. • While transferring 'A' Train SW back to the ocean, the breaker for SW-V-54 (Cooling Tower Pump 110A Discharge Valve) tripped on overcurrent and the valve was found to be mechanically bound in the 60% open position. • Cooling flow to 'A' Train SW loads is nominal at 10,000 gpm. Which of the following accurately describes the Technical Specification implications of this failure? A. The 'A' Train Cooling Tower SW loop is operating and OPERABLE. The 'A' Train Ocean SW loop is OPERABLE because both pump switches were placed in pull-to-lock per the normal operating procedure. B. The 'A' Train Cooling Tower SW loop is operating but INOPERABLE. The 'A' Train Ocean SW loop is INOPERABLE as neither ocean SW pump can be started. C. The 'A' Train Cooling Tower SW loop is operating and OPERABLE. The 'A' Train Ocean SW loop is INOPERABLE as neither ocean SW pump can be started. D. The 'A' Train Cooling Tower SW loop is operating but INOPERABLE. The 'A' Train Ocean SW loop is INOPERABLE because both pump switches were placed in pull-to-lock per the normal operating procedure.
Answer:	B
Justification:	A is incorrect, as the cooling tower SW loop is INOPERABLE with the CT pump discharge valve INOPERABLE. The ocean SW loop is not technically INOPERABLE when pump control switches have been placed in PTL by procedure. No Tech Spec action statement entry is required by procedure. B is correct, the cooling tower SW loop is INOPERABLE and the ocean SW loop is also INOPERABLE because of the control interlock between SW-V-54 and the ocean pumps. With SW-V-54 frozen at 60% open neither ocean pump can be started. C is incorrect because the cooling tower SW loop is INOPERABLE. D is incorrect as ocean pump switch position doesn't make the ocean loop INOPERABLE.
Direct/New/ Modified	New
K/A #:	2.1.12
K/A Values:	4.0
Cognitive Level:	Application (III)
References:	T.S. 3.7.4. 1-NHY-301107 SH AQ3 & AQ4, Interlock between SW-V-54 and ocean pump start logic. Objective L1090I08RO

Question Number:	SRO 19
Question:	A fire in the Train 'A' Electrical Penetration Area has been confirmed by the Fire Brigade, and the control room crew has entered OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION. Prior to initiating the equipment disabling actions identified in the procedure, the 'A' PZR PORV spuriously opens, causing a Safety Injection actuation. What action should the crew take? IAW A. Continue with the actions of OS1200.00 and close the 'A' PORV block valve. B. Transition to E-0, REACTOR TRIP OR SAFETY INJECTION, to deal with the Safety Injection actuation. C. Transition to OS1200.01, SAFE SHUTDOWN AND COOLDOWN FROM THE MAIN CONTROL ROOM. D. Transition to OS1200.02, SAFE SHUTDOWN AND COOLDOWN FROM THE REMOTE SAFE SHUTDOWN FACILITIES.
Answer:	B
Justification:	A, C, & D are incorrect as the transition to E-0 must be made per the direction on the OAS page of OS1200.00 which states: "If a safety injection occurs during the performance of this procedure, than go to E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1."
Direct/New/ Modified	Modified question from 1996 NRC exam
K/A #:	000067AK3.04
K/A Values:	4.1
Cognitive Level:	Memory (I)
References:	OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION, KEY CAUTION

Applicable
B > C

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Question Number:	SRO 20
Question:	Plant conditions: • The control room has been evacuated due to a fire and the remote safe shutdown facilities have been manned. • Remote safe shutdown system lineups have not yet been initiated. • The Local/Remote switch on Bus E-5 for RH-P-8A is in REMOTE. • SSPS in not defeated. • A valid SI signal has just been received. Which of the following describes the response of RH-P-8A? to the SI signal Addition A. The pump will start and remains running until the "SI" signal is reset, at which time the pump will stop. B. The pump will start and remains running until its associated breaker is tripped locally at the switchgear. C. The pump will not automatically start, but the operator can start/stop the pump using the local control switch at the switchgear. D. The pump will not automatically start, nor can it be started locally due to the RMO lockout.
Answer:	B
Justification:	A is incorrect because the 'SI' signal cannot be reset from RSS panel nor would the pump shutdown if only the 'SI' signal were reset. B is correct, per the schematic diagram. C & D are incorrect because the pump will automatically start on an 'SI' signal with the Local/Remote switch in the REMOTE position.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000068AA1.21
K/A Values:	4.1
Cognitive Level:	Memory (I)
References:	RH system text Logic diagram 1-NHY-503761, RH pump logic diagram. Objective L1115I10RO

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Question Number:	SRO 21
Question:	Plant conditions: • A large LOCA has occurred. • The crew is implementing the actions of E-1, LOSS OF REACTOR OR SECONDARY COOLANT • Containment Pressure has increased above 18 psig and the crew has verified proper CBS actuation. • All containment Phase 'A' and 'B' isolation valves indicate closed by status panel with the exception of CC-V168, ('A' Train ORC containment isolation). • The valve position indicating lights for CC-V168 indicate mid position (both Green and Red lamps illuminated). What action should ~~be taken~~ by the crew? IAW A. Continue with the actions of E-1, manually actuate <u>BOTH</u> CBS/P/CVI switches in each train. B. Continue with the actions of E-1, verify all reactor coolant pumps tripped. C. A valid ORANGE path exists on Containment Integrity (Z). The Unit Supervisor should transition to FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE. D. A valid Yellow path exists on Containment Integrity (Z). The Unit Supervisor should transition to FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.
Answer:	C
Justification:	A, B, & D are incorrect since a valid ORANGE path exists on Containment Integrity (Z). A transition should be made to FR-Z.1 where the actions of Step #3 will attempt to close the unisolated valve.
Direct/New/ Modified	New
K/A #:	000069AA1.01
K/A Values:	3.7
Cognitive Level:	Analysis (III)
References:	F-0.5 Containment (Z) FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE MPCS Safety Parameter Display System Functional Description, Rev. 4 Objective L1212I06RO

setpt for CBS?
1) what's in FRZ.1? (ie manually actuate Both CBS/P/C in A)
2) shouldn't RCPs be tripped

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Clarity a) or c)

Question Number:	SRO 22
Question:	Events have occurred resulting in an inadequate core cooling condition. The crew has entered FR-C.1, RESPONSE TO INADEQUATE CORE COOLING. ECCS flow cannot be verified in either train. IAW What action should the crew take to attempt to establish high-pressure injection? A. Start the Positive Displacement Charging Pump and establish flow through CS-FCV-121. B. Start one RCP to collapse any voids in the RCS that restrict ECCS flow. C. Open one PORV to depressurize the RCS and allow ECCS flow. D. Depressurize all intact SGs to facilitate injection of accumulator tanks.
Answer:	A.
Justification:	A is correct per FR-C.1, step 2 RNO. PDP is alternate high-pressure flow in FR-C.1. B is incorrect because starting RCP's will help heat transfer but will not establish injection C is incorrect because opening a PORV with no injection will cause the condition to worsen D is incorrect because accumulator tanks are low head injection
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	E06EK2.1
K/A Values:	3.6/3.8
Cognitive Level:	Application
References:	FR-C.1, step 2 RNO Lesson Plan L1296I, Objective L1206I05RO

Possible if ... RCS ... - Establish ... RCS parameters

~~Follow up~~

Question Number:	SRO 23
Question:	Plant Conditions: • Power is stable at 50% following a setback caused by a trip of the 'B' Main Feedpump. • The letdown activity radiation monitor (RM-6520-1) went into high alarm 45 minutes ago and the Unit Supervisor entered OS1252.01, PROCESS OR EFFLUENT HIGH RADIATION. • A verified chemistry sample indicates that dose equivalent I-131 is 2.75 microcuries per gram. IAW What action should be taken by the crew? A. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. Refer to OS1202.05, RCS HIGH ACTIVITY, after transitioning to ES-0.1, REACTOR TRIP RESPONSE. B. Refer to OS1202.05, RCS HIGH ACTIVITY. Technical Specifications require that the plant be in MODE 3 with Tave less than 500°F within 6 hours. C. Refer to OS1202.05, RCS HIGH ACTIVITY. Technical Specifications allows for continued operation for up to 48 hours under these conditions while increasing RCS sampling requirements to once every 4 hours. D. Refer to OS1202.05, RCS HIGH ACTIVITY, and reduce letdown flow to minimize CVCS contamination.
Answer:	C
Justification:	A is incorrect, there is no immediate plant trip requirement for high RCS activity. B is incorrect, Technical Specifications allow 48 hours of continued operation with dose equivalent, I-131 greater than 1.0 but <180 microcuries/gram dose equiv. iodine at this power level. C is correct, Tech Spec 3.4.8 action a allows continued operation as discussed above and requires RCS sampling frequency of once every 4 hours as long as activity is > 1 microcurie/gram. D is incorrect as OS1202.15 has the operator Increase letdown flow to maximum to cleanup the letdown stream using the demineralizers
Direct/New/ Modified	New
K/A #:	000076AK3.05
K/A Values:	3.6
Cognitive Level:	Analysis (III)
References:	OS1202.05, RCS HIGH ACTIVITY Technical Specification 3.4.8 Objectives L1181I09RO

Question Number:	SRO 24
Question:	A Loss of all AC power has occurred. The crew is performing the actions of ECA-0.0, LOSS OF ALL AC POWER. The crew commences dumping steam from all steam generators to minimize RCS leakage. Which of the following describes the reason that the steam generators should **NOT** be depressurized below 125 psig? A. Remaining above this pressure reduces Pressurized Thermal Shock concern for the reactor vessel. B. It represents the minimum pressure that the steam generators serve as an effective heat sink. C. Minimizes the possibility of SI accumulator nitrogen intrusion into the RCS. D. It represents the minimum pressure that the steam generators can effectively supply the Turbine Driven EFW pump.
Answer:	C.
Justification:	A is incorrect because there is no pressure increase associated with the cooldown. B is incorrect because the steam generators act as a heat sink all the way down to atmospheric pressure. C is correct by ECA-0.0. The pressure should be low enough to inject accumulators, but not so low as to allow nitrogen into the RCS, because the accumulators cannot be isolated electrically. D is incorrect because the TDEFW pump can operate with supply pressure as low as 85 psig (EFW System text)
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000055EK3.02
K/A Values:	4.3/4.6
Cognitive Level:	Memory (I) **NOTE: Re-evaluated from '98 exam (EOP basis question – memory)**
References:	ECA 0.0 and Background Document Lesson Plan L1201I, Objective L1201I03RO

Provide stm tables?
Yes

Question Number:	SRO 25
Question:	The plant is at 81 % power. A pressurizer code safety valve inadvertently lifts. The following indications exist: • Pressurizer pressure 2205 psig and DECREASING. • Temperature downstream of the safety valve indicates 276°F and INCREASING very slowly • PRT pressure 47 psig and INCREASING very slowly Which of the following is the reason for the temperature indication seen downstream of the safety valve? A. The enthalpy ~~temperature~~ of the saturated fluid in the pressurizer vapor space decreases rapidly when it becomes subcooled in the safety valve tailpipe. B. The temperature of the saturated fluid in the pressurizer vapor space decreases as it loses energy due to the high-velocity head loss in the safety valve tailpipe. C. The energy of the saturated fluid in the pressurizer vapor space decreases as it passes through a safety valve, resulting in a temperature indication corresponding to the low-energy fluid in the tailpipe. D. The energy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to the pressure in the PRT.
Answer:	D.
Justification:	A is incorrect because the fluid does not become subcooled in the tailpipe. B is incorrect because energy of the fluid has not changed in the safety valve tailpipe. C is incorrect because the energy (enthalpy) does not go down during a throttling (thru a safety valve) process. D is correct because a leaking valve (throttling process) is a constant enthalpy process. If enthalpy does not change, you can follow the Mollier diagram across (left to right) to the new pressure, and follow the constant pressure line up to the saturation curve. The temperature indicated is the temperature of the vapor at the new pressure
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000008AK3.02
K/A Values:	3.6/4.1
Cognitive Level:	Comprehension (II)
References:	Steam Tables, Mollier Diagram. GFES program – various objectives

Question Number:	SRO 26
Question:	A Small Break LOCA has occurred. The crew is performing the actions of ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. Containment pressure is 5.2 psig and slowly decreasing. ECCS pumps have been stopped. Normal Charging is aligned. The crew is depressurizing the RCS to minimize subcooling. When the depressurization is stopped, the following conditions exist: • RCS Subcooling is 45°F and DECREASING slowly • Pressurizer Level is 62% and DECREASING slowly IAW Based on these indications, what actions should be taken? A. Establish letdown flow to reduce Pressurizer Level to 5%. B. Manually START ECCS pumps as necessary to increase subcooling. C. REINITIATE Safety Injection and verify all safeguards equipment has actuated. D. Continue with the cooldown to cold shutdown. Control charging flow to maintain Pressurizer level greater than 35%.
Answer:	D.
Justification:	A is incorrect, letdown is not established in ES-1.2. B & C are incorrect, subcooling and pressurizer levels are above the criteria for ECCS reinitiation. Safety injection would not be reinitiated. ECCS pumps would be started individually as needed. D is correct, the subcooling and pressurizer level requirements of step #22 are met. The crew should continue the cooldown and control charging flow to maintain pressurizer level.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	000009EA2.01
K/A Values:	4.2/4.8
Cognitive Level:	Application (III)
References:	ES-1.2, step 22 or Operator action summary (ECCS Reinitiation Criteria) Lesson Plan L1204I, Objective L1204I02RO

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Question Number:	SRO 27
Question:	A Plant Trip and Safety Injection has occurred, due to a Steam Generator Fault inside containment. The following conditions exist: • All automatic equipment responds as expected • Containment pressure is 3.2 psig and slowly increasing • RCS pressure is 1750 psig and decreasing • Subcooling margin is 105 degrees F and increasing • Pressurizer level is 22% and decreasing Assuming conditions do not significantly change, in which of the following procedures would you expect to be directed to stop one charging pump? A. In E-2, FAULTED STEAM GENERATOR ISOLATION. B. In E-1, LOSS OF REACTOR OR SECONDARY COOLANT. C. In ES-1.2, POST-LOCA COOLDOWN AND DEPRESSURIZATION. D. In ES-1.1, SI TERMINATION.
Answer:	D
Justification:	A is incorrect, no actions exist in E-2 to shutdown ECCS pumps. B is incorrect, the only ECCS pumps stopped in E-1 are the RHR pumps if RCS pressure is >260# and pressure is stable or increasing. C is incorrect, a transition to ES-1.2 will not be made under conditions given in the stem. D is correct, when the faulted S/G completes blowing down a transition to ES-1.1 will eventually be made from E-1. ES-1.1 directs stoppage of all but one charging pump and re-establishment of normal charging and letdown.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	E02EA2.1 2.4.48
K/A Values:	4.2 3.8
Cognitive Level:	Analysis (III)
References:	E-1, E-2, ES-1.2, ES-1.1 Objectives L1204I03RO

Detailed memorization of procedures?

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Question Number:	SRO 28
Question:	Why does a precaution in OS1008.01, CHEMICAL AND VOLUME CONTROL SYSTEM MAKEUP OPERATIONS, warn the operator to closely monitor VCT level if CS-LT-112 fails high? A. Automatic makeup is defeated. B. Divert on high VCT level is defeated. C. Makeup will **NOT** terminate automatically. D. Swapover to the RWST will **NOT** occur on SI actuation.
Answer:	A
Justification:	LT-112 failing high results in full divert (B incorrect). VCT level will decrease, makeup will fail to initiate. (A correct, C incorrect). Swapover on Low-Low level will not occur, however, SI will still cause swapover. (D incorrect.)
Direct/New/ Modified	Direct from bank
K/A #:	000022AK3.02
K/A Values:	3.5/3.8
Cognitive Level:	Comprehension (II)
References:	OS1008.01 Precaution, 1-CS-D20725, 1-NHY-506725 Objective L1105I05RO

?

Question Number:	SRO 29
Question:	During reduced inventory conditions a hot leg vent path capability must be established and maintained to ______________. A. allow a vent path for nitrogen. B. prevent pressurizer surge line flooding. C. limit RCS pressurization in the event of a loss of RHR cooling D. ensure that ultrasonic level instrumentation reads accurately.
Answer:	C
Justification:	A is incorrect, the PZR is vented to atmosphere and therefore contains air along with a good portion of the RCS, nitrogen venting is not an issue under these conditions. B is incorrect, the pressurizer serves as the hot leg vent path. Vent would not prevent water from entering PZR if water level expansion occurred due to RCS heatup. C is correct as stated in OS1000.12 (Caution prior to prerequisite 3.1). D is incorrect, hot leg vent path has no effect on accuracy of loop ultrasonic level instrumentation.
Direct/New/ Modified	New (Modified from V.C summer SRO exam)
K/A #:	000025AA1.02
K/A Values:	3.8/3.9
Cognitive Level:	Comprehension (II)
References:	OS1000.12 Objective L1705I11RO

Question Number:	SRO 30
Question:	Due to a failure of PZR pressure channel 455, PZR pressure channels 457/456 have been selected for control and backup respectively. Sometime later, channel 457 fails LOW. Which of the following describes the effect, if any, this failure has on PORV operation in the present mode? A. Only PORV 456A is prevented from opening automatically. B. Only PORV 456B is prevented from opening automatically. C. Both PORVs are prevented from opening automatically. D. Both PORVs will open automatically when required.
Answer:	C
Justification:	C is correct in accordance with the note prior to Step #1 of OS1201.06, "PZR Pressure Instrument PT 455/458 Failure." A,B, & D are incorrect in accordance with the process control block diagram.
Direct/New/ Modified	Modified from facility Exam Bank
K/A #:	000027AA1.01
K/A Values:	3.9
Cognitive Level:	Analysis (III)
References:	OS1201.06, PZR PRESSURE INSTRUMENT PT 455/458 FAILURE. (Note prior to step #1) 1-NHY-509026, PZR pressure control process control block diagram. Objective L1182I14RO

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Question Number:	SRO 31
Question:	Plant conditions: • A reactor startup was being performed when source range channel N31 indication failed. • The startup was halted in accordance with the requirements of Technical Specifications. • Source range channel N32 power is stable at $6x10^3$ CPS • The level trip bypass switch for channel N31 is in the BYPASS position • During performance of trouble shooting on channel N31 its control power fuses blow. What is the expected plant response, and why? A. The reactor will NOT trip. Source rang channel N31 is in level trip bypass. B. The reactor will trip. Level trip bypass requires control power to function and the N31 high flux trip bistable is de-energized on loss of control power. C. The reactor will NOT trip as the trip signal requires control power to function. D. The reactor will trip. Loss of control power de-energizes the backup trip bistable, which is in the instrument power circuit.
Answer:	B
Justification:	B is correct, loss of control power trips (de-energizes) the SR high flux trip bistable for N31. Control power is required for the level trip bypass function to operate. Therefore the trip condition will be passed on to SSPS and the RX will trip. A & C are incorrect as they state the Rx will NOT trip. D is incorrect because the reason given for the cause of the reactor trip relies on a component which does not exist.
Direct/New/ Modified	New
K/A #:	000032AK2.01
K/A Values:	2.7/3.1
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1112I, objective L1112I08RO & L1112I09RO

Plausible or trick Q?

Question Number:	SRO 32
Question:	During a Reactor Startup the following conditions exist: • P-6 has just energized • Source Range Channel N-31 indicates 5 X 10^3 CPS • Source Range Channel N-32 indicates 4 X 10^3 CPS • Intermediate Range Channel N35 indicates 2 X 10^{-10} amps • Intermediate Range Channel N36 indicates 2 X 10^{-11} amps Which of the following is the likely cause of the above readings? A. Intermediate Range Channel N35 is undercompensated. B. Intermediate Range Channel N36 is undercompensated. C. Intermediate Range Channel N35 is overcompensated. D. Intermediate Range Channel N36 is overcompensated.
Answer:	A.
Justification:	A is correct because an undercompensated NI will read higher than normal. Additionally, the lesson plan states that P-6 energizes at approximately 1E-10 amps on 1 of 2 IR channels, which corresponds to approximately 4X10^4 CPS on the SR channels. B is incorrect because N36 would read higher if it were undercompensated C is incorrect because N35 would read lower if it were overcompensated D is incorrect because N36 is reading consistent with the expected overlap with the 2 source range channels
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000033AA2.11
K/A Values:	3.1/3.4
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1112I, Objective L1112I06RO

Question Number:	SRO 33
Question:	Plant conditions: • Stable at 100% power. • 'C' main steamline radiation monitor is in "ALERT" (Yellow) alarm. • The crew has entered OS1227.02, STEAM GENERATOR TUBE LEAK • Successive chemistry samples indicate a leakage rate of 80 gallons per day (gpd) and the rate of change in the leak rate is estimated at less than 5 gpd/hr. What action is required in accordance with the Abnormal procedure? A. No shutdown required. B. Shutdown to MODE 3 within 3 hours. C. Shutdown to MODE 3 within 6 hours. D. Shutdown to MODE 3 within 8 hours.
Answer:	A
Justification:	A is correct, a plant shutdown is not required if leakage is less than 150 gpd and the increase in leak rate is less than 30 gpd/hr. B, C, & D are all incorrect as they represent various shutdown rates in the procedure for leakage rates greater than 150 gpd.
Direct/New/ Modified	New
K/A #:	000037AK3.05
K/A Values:	4.0
Cognitive Level:	Analysis (III)
References:	OS1227.02, STEAM GENERATOR TUBE LEAK Objective L1190I02RO

Question Number:	SRO 34
Question:	The plant has sustained a Steam Generator tube rupture concurrent with a loss of Off-Site power. All safeguards systems functioned as designed. Actions of E-3, STEAM GENERATOR TUBE RUPTURE, have been performed. The crew is preparing to cool down and depressurize the RCS to MODE 5. Based on current plant conditions, which of the following cooldown methods is preferred? A. ES-3.1, POST SGTR COOLDOWN USING BACKFILL, because it minimizes radiological release. B. ES-3.2, POST SGTR COOLDOWN USING BLOWDOWN, because it minimizes the spread of contamination to secondary plant components. C. ES-3.3, POST SGTR COOLDOWN USING STEAM DUMP, because it is the fastest method of cooldown. D. ES-3.3, POST SGTR COOLDOWN USING STEAM DUMP, because it conserves CST inventory.
Answer:	A.
Justification:	A is correct with the correct qualifier, IAW E-3 background document, page 44. B is incorrect because the qualifier is incorrect. Blowdown can possibly spread contamination. C and D are incorrect because Condenser Steam dumps are not available, leaving ASDVs which are the least desirable because of radiological concerns
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	000038EA2.08
K/A Values:	3.8/4.4
Cognitive Level:	Comprehension (II)
References:	ERG background; E-3 series, page 44 Lesson Plan L1205I, Objective L1205I03RO

Question Number:	SRO 35
Question:	The plant is at 15% power when a total loss of Main Feedwater occurs. The reactor does not trip, and the crew enters FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS. What function, if any, will the ATWS Mitigation System provide under these conditions? A. The ATWS Mitigation System is not armed under these conditions. B. The ATWS Mitigation System will send a start signal to the EFW pumps when 1/4 SG NR levels are less than 5%. C. The ATWS Mitigation System will send a trip signal to the Main Turbine when 2/4 detectors on 1/4 SGs are less than 14%. D. The ATWS Mitigation System will send a start signal to the EFW pumps when 3/4 SG NR levels are less than 5%.
Answer:	A
Justification:	A is correct because the system is armed above 20% power. B is incorrect because the correct actuation logic requires NR level to be below 5% in ¾ SGs. C is incorrect, this is the SSPS actuation logic for the EFW pumps. D is incorrect, while this answer defines the proper actuation logic the ATWS mitigation system will not function because it is below it's arming power level.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000054AA2.03
K/A Values:	4.2
Cognitive Level:	Comprehension (II)
References:	EFW Detailed System Description, Lesson Plan 1127, OBJ L1127I03RO

Question Number:	SRO 36
Question:	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 a valid RED path condition on the heat sink CSF. When the crew checked whether heat sink was required, the Primary Operator reported that RCS pressure was 700 psig and slowly decreasing. The Secondary Operator reported that all S/G pressures were approximately 950 psig and stable. Based on this information, the Unit Supervisor transitioned to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 1. Which of the following summarizes plant conditions? A. A LOCA is in progress and heat removal rate due to break flow plus SI injection flow is GREATER than the decay heat rate, therefore, a secondary heat sink is not required. B. A LOCA is in progress and heat removal rate due to break flow plus SI injection flow is LESS than the decay heat rate, therefore, a secondary heat sink is not required. C. A LOCA is in progress and with primary pressure less than secondary pressure, heat must be transferred to the S/Gs therefore, a return to E-1 is made to restore Steam Generator levels. D. A LOCA is in progress and with primary pressure less than secondary pressure, heat cannot be transferred to the S/Gs, therefore, a return to E-1 is made to depressurize all S/Gs prior to returning to FR-H.1.
Answer:	A
Justification:	A is correct based on conditions given in the stem, S/Gs are not required as a heat sink. B is incorrect as it states the opposite condition from the correct answer. C is incorrect as FR-H.1 would be the appropriate procedure for restoration of S/G levels and a heat sink is not required. D is incorrect, with conditions in stem a transition to ES-1.2 would be made prior to S/G depressurization step in E-1 (step 15).
Direct/New/ Modified	New
K/A #:	E05EK2.2
K/A Values:	4.2
Cognitive Level:	Analysis (III)
References:	OBJ L1211I03RO FR-H.1 background document.

Backwards logic

Perhaps - why did the crew exit FR-H.1

Question Number:	SRO 37
Question:	Which of the following will occur on a loss of Vital DC Bus 11A? A. Both EFW pumps start and the MFRV and bypass valves fail open. B. 'A' train PCCW temperature control and bypass valves fail to their minimum cooling positions (HX bypass). C. The 'A' train P-14 solenoids on the MFRV and MFRV bypass valves are de-energized causing these valves to fail closed. D. The 'A' train P-12 solenoids on the steam dump valves are de-energized causing the steam dumps to fail open.
Answer:	C
Justification:	A is incorrect because the MFRV and MFRV bypass valves will fail closed not open. B is incorrect because PCCW temperature control fails to the full cooling position. C is correct in accordance with the caution prior to Step #1 of OS1248.01. D is incorrect because the P-12 solenoids result in a loss of steam dump capability.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000058AK3.02
K/A Values:	4.2
Cognitive Level:	Memory (I)
References:	OS1248.01, Loss of Vital 125 VDC Bus, Caution Electrical Drawings Objectives L1186I10RO

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Question Number:	SRO 38
Question:	The following event has occurred: • SGTR on 'B' SG • One safety valve on 'B' SG is stuck open • RCS activity is high due to failed fuel. spell out • TDEFW pump steam supply from 'B' SG has not been isolated. An initial ODPS run on the ruptured/faulted SG resulted in a Site Area Emergency classification with PAR group 'A'. A field monitoring team has recommended that another ODPS run be performed because the TDEFW pump is running. What is the correct answer? Which of the following should be performed? context of question IAW for should A. Another ODPS run is NOT required since the release is from the ruptured/faulted SG. B. Run the ODPS program again using the same pathway and current SG pressure. If PAR group 'B' is indicated, reclassification is not required. C. Run the ODPS program again using the UNMONITORED pathway. If PAR group 'B' is indicated reclassification is not required. D. Run the ODPS program again using the UNMONITORED pathway. If PAR group 'B' is indicated, reclassify the event. C
Answer:	D
Justification:	D is correct because unmonitored is the pathway to be used for the TDEFW pump running and PAR group 'B' is more conservative than group 'A'. A is incorrect because another ODPS run is required. B & D are incorrect because the pathway is wrong and PAR 'B' is more conservative than PAR 'A'.
Direct/New/ Modified	New
K/A #:	000060AK1.04
K/A Values:	3.7
Cognitive Level:	Application (III)
References:	ER-5.7, L3063I14SR

Check revised

Question Number:	SRO 39
Question:	The plant is at full power. A report from the manufacturer of the Containment – Post LOCA – Area Monitors, R-6576A and R-6576B (1AM106, 1AM107), identifies a common problem with the radiation monitors' power supplies and both are declared INOPERABLE. what is max time Referring to the attached Technical Specifications, ~~how long can~~ the unit remain at full power with the monitors INOPERABLE and comply with the applicable ACTION statement? A. 1 hour B. 72 hours C. 7 Days D. Indefinitely
Answer:	D
Justification:	Technical Specification 3.0.3 does not apply (A incorrect). Preplanned alternate methods of monitoring containment must be initiated within 72 hours and the monitors returned to service within 7 days, or, a special report to the Commission pursuant to Specification 6.8.2 submitted within 14 days outlining actions taken, cause of inoperability and plans to schedule for restoring system to OPERABLE status. (No shutdown is required, D is correct.)
Direct/New/ Modified	Direct from bank
K/A #:	000061AA2.06
K/A Values:	4.1
Cognitive Level:	Application (III)
References:	OS1252.03, Tech Spec 3.3.3.1 ACTION 27 Objective L1140I02RO

too encompassing

PINK PAGE

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

5A4 91

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39a

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment - Post LOCA - Area Monitor	1	2	All	≤ 10 R/h	27
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26#
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26#
2. Containment Ventilation Isolation					
a. On Line Purge Monitor	1	2	1, 2, 3, 4	*	23
b. Manipulator Crane Area Monitor	1	2	6##	**	23
3. Main Steam Line	1/steam line	1/steam line	1, 2, 3, 4	N.A.	27
4. Fuel Storage Pool Areas					
a. Fuel Storage Building Exhaust Monitor	N.A.	1	***	****	25
5. Control Room Isolation					
a. Air Intake-Radiation Level					
1) East Air Intake	1/intake	2/intake	All	****	24
2) West Air Intake	1/intake	2/intake	All	****	24
6. Primary Component Cooling Water					
a. Loop A	1	1	All	< 2 x Background	28
b. Loop B	1	1	All	< 2 x Background	28

TABLE NOTATIONS

* Two times background; purge rate will be verified to ensure compliance with Specification 3.11.2.1 requirements.
** Two times background or 15 mR/hr, whichever is greater.
*** With irradiated fuel in the fuel storage pool areas.
**** Two times background or 100 CPM, whichever is greater.
\# The provisions of Specification 3.0.4 are not applicable.
\## During CORE ALTERATIONS or movements of irradiated fuel within the containment.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 23 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment ventilation isolation valves are maintained closed.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

ACTION 25 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, collect grab samples daily from the Primary Component Cooling Water System and the Service Water System and analyze the radioactivity until the inoperable Channel(s) is restored to OPERABLE status.

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment - Post LOCA - Area Monitor	S	R	Q	All
b. RCS Leakage Detection				
1) Particulate Radioactivity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Containment Ventilation Isolation				
a. On Line Purge Monitor	S	R	Q	1, 2, 3, 4
b. Manipulator Crane Area Monitor	S	R	Q	6#
3. Main Steam Line	S	R	Q	1, 2, 3, 4
4. Fuel Storage Pool Areas				
a. Radioactivity-High-Gaseous Radioactivity	S	R	Q	*
5. Control Room Isolation				
a. Air Intake Radiation Level				
1) East Air Intake	S	R	Q	All
2) West Air Intake	S	R	Q	All
6. Primary Component Cooling Water				
a. Loop A	S	R	Q	All
b. Loop B	S	R	Q	All

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.
# During CORE ALTERNATIONS or movement of irradiated fuel within the containment.

PINK PAGE

Technical Clarification

SECTION I - REQUEST FOR CLARIFICATION		
Originator: J. Connolly (Rev. 01)		Date: 02/01/99
Technical Clarification Title: COP Monitor Setpoints		
Technical Clarification No.: TS-091 (Rev.01)		
Type of Clarification:	Tech Spec (TS) ☒	Licensing (LS) ☐

REQUEST FOR CLARIFICATION: (Attempt to state the request as a question.)

The COP monitor setpoints are required to be set at "less than or equal to" 2X background. To determine this setpoint, flow must be present in the COP system.

Is it permissible to open the purge supply and exhaust isolation valves, prior to determining this setpoint, for the purpose of determining the background?

CONCURRENCE: ______________________ Group Manager   2/11/99 Date

SECTION II - INITIATION

RECEIVED BY REGULATORY COMPLIANCE: ______________________ Regulatory Compliance Supervisor   2/11/99 Date

(Continued)

Technical Clarification

SECTION III -EVALUATION

The setpoint for the COP monitors is required to be set at less than or equal to 2X background (less than or equal to 2X the value at the start of purging). To determine this value, the COP system must be in operation. It is therefore permissible to open the purge supply and exhaust isolation valves and initiate system operation (for a period of up to 1 hour) prior to determining this setpoint, provided that actions to determine the setpoint are pursued without delay and in a controlled manner, and that during this period, the monitors remain in operation utilizing a setpoint which will provide system actuation if required.

Another acceptable method to determine the initial setpoint is described in Health Physics Study / Technical Information Document (HPSTID) 98-007 "Containment On-line Purge (COP) Radiation Monitor Response to Noble Gases" which can be used to provide a method to estimate the response of the Containment On-Line Purge (COP) Radiation Monitors (RM-6527A and RM-6527B) to concentrations of routinely purged noble gasses from containment. An estimated response to the gas concentration in containment can be combined with the pre-purge (ambient) background value of the monitors to determine the monitor setpoints prior to initiating purging operations.

The initial setpoint should be established by Operations and Health Physics personnel.

REFERENCE: T.S. 3.3.2, Table 3.3-3, Item 3.c.4

T.S. 3.3.3.1, Table 3.3-6, Item 2.a

Prepared By: James W. Connolly 2/16/99 (Date)

Concurrence: 2/16/99 Cognizant Group Manager (Date)

SECTION IV - REVIEW AND APPROVAL

(Check Appropriate Boxes)

☒ 2/16/99 Regulatory Compliance Manager (Date) ☐ (Date)

☒ Station Director (Date) ☐ 2/17/99 (Date)

☒ (Date) ☐ (Date)

☐ (Date)

SORC MEETING NO.: 99-017 DATE: 2/17/99

39f

Technical Clarification
TS-091 (Rev. 01)

Background:

The Containment On-Line Purge (COP) system is primarily used to purge the containment atmosphere periodically during plant operation and to reduce the airborne activity levels in the containment. The COP system replenishes the containment structure atmosphere with fresh outside air and directs potentially contaminated air to the normal exhaust air cleaning unit in the Primary Auxiliary Building. The COP system containment isolation valves (COP-V1, COP-V2, COP-V3 and COP-V4) satisfy containment isolation requirements by automatically closing when a containment ventilation isolation signal (CVIS) is generated when any one of the following conditions occurs:

- An "S" signal is generated,
- Containment Spray is manually initiated,
- Phase "A" isolation is manually initiated, or
- High Radiation is detected by the COP exhaust line radiation monitors (RM-6527A, or RM-6527B).

The COP system exhaust line radiation monitors monitor the air quality of the containment building via the containment purge exhaust. A two-of-two detector logic is used, where by both detectors on a per train basis, must be in high alarm to provide a CVIS. During COP system operation, the radiation monitor trip setpoint is required to be set at less than two times the background radiation levels as required by Technical Specification Table 3.3-4.

In order to reliably set the alert and high setpoints for the COP exhaust line radiation monitors, it is necessary to establish air flow through the COP system exhaust lines. It has been past practice (per OS1023.69 "Containment On-Line Purge System Operation") to establish the alert and high setpoints at 1×10^5 Counts Per Minute (CPM) until the background levels for the COP exhaust line radiation monitors could be established. This level has been considered conservative enough to provide system isolation if required during the system start-up period (less than 1 hour of operation). Once background levels are determined, the alert and high alarm setpoints can be calculated and the setpoints adjusted as appropriate to meet the requirements of the Technical Specifications.

In order to provide a more realistic setpoint for the generation of a CVIS prior to the start-up of the COP system, a Health Physics Study / Technical Information Document (HPSTID 98-007) "Containment On-line Purge (COP) Radiation Response to Noble Gases" was developed by the Health Physics Department. This HPSTID provides a method to estimate the response of the Containment On-Line Purge (COP) Radiation Monitors (RM-6527A and RM-6527B) to concentrations of routinely purged noble gasses from containment. An estimated response to the gas concentration in containment can be combined with the pre-purge (ambient) background value to determine the monitor setpoint prior to initiating purging operations.

TECHNICAL CLARIFICATION

*** * * * SECTION I - REQUEST FOR CLARIFICATION * * * ***

Originator: W. NICHOLS Date: 3/25/91

Technical Clarification No.: TS-142

Type of Clarification:

TechSpec ☒ FSAR(Excluding 17.2) ☐ FSAR 17.2 ☐ Other Licensing ☐

DESCRIPTION OF ISSUE: (Include research information and identified differences that require resolution.)

Technical Specifications 4.3.2.1, 4.3.3.1, 4.3.3.9 and 4.3.3.10 require periodic surveillance testing of the Radiation Data Management System (RDMS). Part of the surveillance testing requires the performance of a Digital Channel Operational Test (DCOT) on each channel.

During the performance of the DCOT the alarm/trip setpoint is reduced below the background radiation level in order to verify alarms and any associated trips & interlocks actuate properly.

Does lowering of the channel setpoint below the background radiation level ALARM/TRIP SETPOINT values stated in T.S. Table 3.3-6 cause the channel to be inoperable?

CONCURRENCE: [signature] 4/15/91

Group Manager (Date)

*** * * * SECTION II - INITIATION * * * ***

RECEIVED REGULATORY COMPLIANCE: [signature] 3/27/91

Lead Engineer - Compliance (Date)

GROUP MANAGER ASSIGNED: JM Peschel

(Name/Dept) (Date)

*** * * * SECTION III - EVALUATION * * * ***

ANSWER: NO.

The temporary lowering of a RDMS channel setpoint, by RDMS data base manipulation to verify alarm/trip functions, does not prevent the channel from continuously monitoring radiation levels (except WRGM). Additionally, when the setpoint is lowered below background radiation levels the associated trip functions will actuate equipment in their required operating mode as if a high radiation condition exists. The channel remains OPERABLE because monitoring and associated trip functions are not inhibited.

Therefore, during performance of a RDMS channel DCOT, the LCO remains satisfied. Entering an ACTION statement is not appropriate nor required (except for WRGM DCOT). However, because the channel is in alarm status, increased operator vigilance is required to note any increase in radiation levels during the DCOT surveillance period and to take remedial actions if required.

See attached background information, particularly for WRGM DCOT and ACTION statement applicability.

Prepared By: Reset Leclerc 3/29/91 (Date)

Concurrence: J M Ferrebee 4/10/91 Cognizant Group Manager (Date)

*** * * * SECTION IV - REVIEW AND APPROVAL * * * ***

Check Appropriate Boxes

☒ J M Ferrebee 4/10/91 Regulatory Compliance Manager (Date)	☒ 4/15/91 Maintenance Manager (Date)
☒ Joseph M Griffith April 17 1991 Operations Manager (Date)	☒ 3/27/91 Lead Engineer - Compliance (Date)
☐ Chemistry and Health Physics Manager (Date)	☒ 5/8/91 Station Manager (Date)
☒ M Kline 4/23/91 Technical Support Manager (Date)	☐ (Date)

SORC MEETING NO.: 91-76 DATE: 5/8/91

APPROVED BY: 5/14/91 Executive Director - Nuclear Production (Date)

BACKGROUND INFORMATION CONCERNING RADIATION MONITORING DATA SYSTEM CHANNEL OPERABILITY DURING DCOT SURVEILLANCES

The Radiation Data Management System (RDMS) is a digital computer-based radiation monitoring system. The system consists of multiple channels for monitoring radiation levels in designated areas and process streams. When radiation levels exceed a pre-determined setpoint, the RDMS will initiate alarms and/or trip actuations.

Technical Specifications 4.3.2.1, 4.3.3.1, 4.3.3.9 and 4.3.3.10 require periodic surveillance testing of channels associated with the Radiation Data Management System (RDMS). Part of the surveillance testing requires the performance of a Digital Channel Operational Test (DCOT) on each channel designated within the Technical Specifications.

The DCOT is performed by manipulating the database to temporarily lower the setpoint on the channel being tested to a value which is lower than the radiation level (background) being sensed by the monitor. Lowering the setpoint provides the requisite verification that the channel alarms, trips and/or interlocks are operational. Upon completion of the DCOT the channel setpoint is returned to its required setpoint as specified by the Technical Specifications.

When radiation is detected each RDMS channel conditions the signal from its detector(s) for proper input to an RM-80 microprocessor. The microprocessor mathematically manipulates the conditioned signal within the microprocessor data base to arrive at a radiation value. The data base (i.e., radiation value) is then continuously compared to a selected digital setpoint value within the microprocessor. When the radiation value is higher than the setpoint value the microprocessor executes additional instructions to a control system to perform alarming and/or tripping functions. Since the setpoint is a digital value within the microprocessor data base, the setpoint can be altered by operator manipulation of the data base. By intentionally lowering a RDMS channel setpoint to below the continuously updated radiation value within the data base the alarm and/or trip functions can be verified. Therefore, the temporary lowering of the setpoint does not inhibit the channel from continuously monitoring radiation levels (except WRGM) or performing any associated trip functions. Additionally, when the setpoint is lowered below background radiation levels the associated trip functions will actuate equipment in their required "safe" operating mode as if a high radiation condition (i.e., above the setpoint) exists.

From the above statements, the RDMS channel(s) remains OPERABLE because monitoring and associated trip functions are not inhibited when setpoint changes are made. Therefore, during performance of a RDMS channel DCOT (except WRGM) the RDMS channel remains OPERABLE.

The ALARM/TRIP SETPOINT values given in T.S. Table 3.3-6 are maximum values that specific RDMS channel setpoints can be adjusted. Therefore, temporarily lowering the channel setpoint(s) below the ALARM/TRIP SETPOINT value is a temporary adjustment in the conservative direction (i.e., the alarm/trip functions occur sooner) and; coupled with the above statement that the channel remains OPERABLE even when lowering the setpoint, the Limiting Condition for Operation (LCO) for the channel under test remains satisfied. Thus, entering an ACTION statement during a DCOT for RDMS channels is not appropriate nor required, except when a WRGM DCOT is performed.

The WRGM consists of two flow paths with three detection channels for low, medium and high range radiation monitoring of the plant vent stack. During normal operation, the low range (high flow) path is used and the mid/high range (low flow) path is shut down. As activity increases above a pre-determined setpoint, the mid/high range path is automatically placed into operation. If activity levels continue to increase through the mid range, the low range path is automatically isolated and purged (to minimize activity buildup in the sample skid). During the WRGM DCOT, the low range detector path, which monitors normal radiation background levels, will become inoperable in order to test the mid/high range channel functions. Therefore, during the WRGM DCOT the appropriate ACTION statement must be entered until the low range path is placed back in service.

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It should be noted that whenever the setpoint is lowered below the background radiation value the channel will be in alarm status. Therefore, increased operator vigilance is required to note any increase in radiation levels during the DCOT surveillance period and to take remedial actions if required.

RNL

TECHNICAL CLARIFICATION

*** * * * SECTION I - REQUEST FOR CLARIFICATION * * * ***

Originator: R. C. Sterrett Date: 5/20/91 NOT IMPORTANT WHO 7/24/91

Technical Clarification No.: TS-149

Type of Clarification:

TechSpec [X] FSAR(Excluding 17.2) [ ] FSAR 17.2 [ ] Other Licensing [ ]

DESCRIPTION OF ISSUE: (Include research information and identified differences that require resolution.)

Tech Spec 3/4 3.3 and Table 3.3-6 Require the Manipulator Crane Area Radiation Monitor to be in operation in mode 6, during core alterations or movements of irradiated fuel within the containment. This Spec. also requires an alarm set point of 15 mR/hr or 2 times background, whichever is greater. Definition 1.9 describes a core alteration as the movement or manipulation of any "component" within the reactor pressure vessel with the head removed and fuel in the vessel. NHY Technical Interpretation TI-004 further clarifies "component" to mean any material that could alter core reactivity to significantly reduce shutdown margin or posses sufficient mass to possibly challenge fuel integrity if mishandled.

Prior to head removal, dose rates at the detector location will be low - perhaps <5 mR/hr, and the setpoint of 15 mR/hr will apply. During head removal, the upper internals will be exposed, and the dose rates at the detector location will rise to an unpredictable level (estimate is 100 mR/hr). As the cavity is flooded and covers the upper internals, dose rates are expected to return to <5mR/hr.

1. Can the radiation monitor be taken out of service during head removal? (Failure to do so would trip the alarm, actuating containment isolation. It is beneficial to have containment purge in operation during this operation to control containment airborne radioactivity.)
2. If the monitor can be removed from service during head removal, would it have to be placed back in service during cavity flooding (due to potential reactivity changes)?
3. During upper internals removal, we may see a similar increase in dose rates. Is upper internals movement considered a core alteration? If it is considered a core alteration, can the alarm set point be set to expected values based on the exposed upper internals condition?
4. Following core re-load, is the monitor required to be in service during cavity draindown? (Upper internals will again be exposed, increasing dose rates above the 15 mR/hr set point) If it is, can the alarm set point be set to the values expected based on initial draindown?

CONCURRENCE: Joseph J. Rafalowski 5/23/91
Group Manager (Date)

*** * * * SECTION II - INITIATION * * * ***

RECEIVED REGULATORY COMPLIANCE: JM Peschel for ATL 5/24/91
Lead Engineer - Compliance (Date)

GROUP MANAGER ASSIGNED: J. M. Peschel
(Name/Dept) (Date)

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*** * * * SECTION III - EVALUATION * * * ***

ANSWER:

1) The reactor vessel head is not an internal component within the reactor vessel and as such, its removal is not considered a CORE ALTERATION. Therefore, the Manipulator Crane Area Radiation Monitor may be disabled during reactor vessel head removal operations.

2) Flooding of the reactor cavity with borated water from the RWST that is within Technical Specification boron concentration limits is not considered an evolution that could alter core reactivity to significantly reduce shutdown margin. Therefore, the Manipulator Crane Area Radiation Monitor is not required to be in service during reactor cavity flooding.

3) During reactor cavity drain down, it would be prudent to have the Manipulator Crane Area Radiation Monitor inservice. The setpoint on the monitor may be established based on the expected radiation levels which were previously experienced during head removal and cavity fill.

4) The upper internals package is an internal component within the reactor vessel. Its removal and reinstallation is considered a CORE ALTERATION whenever fuel is in the reactor vessel and the upper internals package

(Continued on next page)

Prepared By: [signature] 7/23/91 (Date) Concurrence: [signature] 7/23/91 Cognizant Group Manager (Date)

*** * * * SECTION IV - REVIEW AND APPROVAL * * * ***

Check Appropriate Boxes

☐ [signature] 7/23/91 Regulatory Compliance Manager (Date)	☐ Maintenance Manager (Date)
☐ Joseph M Kelly 7/23/91 Operations Manager (Date)	☐ [signature] Lead Engineer - Compliance (Date)
☐ [signature] 7/23/91 Chemistry and Health Physics Manager (Date)	☐ [signature] 7/24/91 Station Manager (Date)
☐ [signature] 7/23/91 Technical Support Manager (Date)	☐ J Rafalowski 7/23/91 (Date)

SORC MEETING NO.: 91-118 DATE: 7-24-91

APPROVED BY: [signature] 8/1/91 Executive Director - Nuclear Production (Date)

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is not fully seated within the reactor vessel, <u>and</u> while the upper internals package is directly above the fuel assemblies. Thus, the Manipulator Crane Area Radiation Monitor must be in service during movement of the upper internals package until the upper internals package is no longer above the fuel assemblies.

5) Technical Specification 3/4.3.3 allows the setpoint to be adjusted to 15 mR/hr or 2 times above background whichever is greater. Thus, the setpoint may be adjusted to expected radiation values based on previous experience. Furthermore, the setpoint may be adjusted at different values (based on previous experience) for various evolutions which are known to significantly alter surrounding background radiation levels, provided the setpoint changes are procedurally controlled to coincide with the evolutions in progress.

BACKGROUND INFORMATION CONCERNING DISABLEMENT OF MANIPULATOR CRANE AREA RADIATION MONITOR DURING REFUELING ACTIVITIES

The Health Physics Department requested a technical clarification of Technical Specification 3.3.3.1, Radiation Monitoring for Plant Operations, as to what refueling operations are considered a CORE ALTERATION and whether the Containment Manipulator Crane Area Monitor High Radiation signal to the Engineered Safeguards Features Actuation System (ESFAS) can be disabled or setpoints readjusted to prevent automatic actuation/isolation of containment ventilation systems during refueling evolutions which are known to significantly alter background radiation levels.

During refueling operations the Containment Manipulator Crane Area Monitor-Channels 6535 A and B is in service to monitor general background radiation levels within the containment building. In the event of a fuel handling accident, these detector channels in conjunction with safeguards actuation signals will isolate the containment online and offline purge isolation valves, and trip the containment pre-entry, refueling supply and containment online purge fans when background radiation levels exceed the predetermined setpoint. However, certain refueling activities such as reactor head removal, reactor cavity filling and draining evolutions, etc., can significantly alter general background radiation levels which can exceed the setpoints of the radiation monitoring instrument channels.

CORE ALTERATION is defined in the Technical Specifications as the movement or manipulation of any component <u>within</u> the reactor pressure vessel with the vessel head removed and fuel in the vessel. NHY Technical Interpretation TI-004 was issued to clarify the word "component" as any material that could alter core reactivity to significantly reduce shutdown margin or possesses sufficient mass to possibly challenge fuel integrity if mishandled. It can be inferred from the definition and technical interpretation that a component possessing sufficient mass within the reactor vessel only would be considered a CORE ALTERATION.

Removal of the reactor vessel head is not considered a CORE ALTERATION since the reactor vessel head is not an internal component within the reactor vessel. It is recognized that the reactor vessel head does have sufficient mass which could challenge fuel integrity if mishandled, however, per NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, a drop of the reactor vessel head could impact the reactor vessel flange, but not directly challenge fuel integrity (inferred), and potentially damage the reactor vessel itself and lead to uncovering the fuel if sufficient leakage resulted beyond water makeup capability. Since dropping of the reactor vessel head would not crush the core, manipulation of the reactor vessel head is not considered a CORE ALTERATION, therefore, the Containment Manipulator Crane Area Monitor may be disabled during removal of the reactor vessel head. In addition, per the definition, CORE ALTERATION can only occur <u>after</u> the reactor vessel head is removed.

The upper internals package is an internal component within the reactor vessel that possesses sufficient mass that could challenge fuel integrity if mishandled. Additionally, past events in the industry as documented in NRC Information Notice 90-77: Inadvertent Removal of Fuel Assemblies From the Reactor Core, identify events where fuel assemblies were attached to the

upper internals package during removal operations of the upper internals package. Any inadvertent removal of a fuel assembly would constitute a change in core reactivity.

Therefore, whenever the upper internals package is not fully seated within the reactor vessel and while the upper internals package is directly above the fuel assemblies the Containment Manipulator Crane Area Monitor must be in service during movement of the upper internals package until the upper internals package is no longer above the fuel assemblies.

Flooding of the reactor cavity with borated water from the refueling water storage tank (RWST) that is within Technical Specification boron concentration limits is not considered an evolution that could alter core reactivity to significantly reduce shutdown margin. Therefore, the Containment Manipulator Crane Area Monitor may be disabled during reactor cavity flooding.

During reactor cavity drain down, it would be prudent to have the Manipulator Crane Area Radiation Monitor inservice. The setpoint on the monitor may be established based on the expected radiation levels which were previously experienced during head removal and cavity fill.

Technical Specification 3.3.3.1 Table 3.3-6 allows the Containment Manipulator Crane Area Monitor setpoint to be adjusted to 15 mR/hr or 2 times above background whichever is greater. Thus, the setpoint may be adjusted to expected radiation values based on previous experience. . Furthermore, the setpoint may be adjusted at different values (based on experience gained) for various evolutions which are known to significantly alter surrounding background radiation levels, provided the setpoint changes are procedurally controlled to coincide with the evolutions in progress.

TECHNICAL CLARIFICATION

SECTION I - REQUEST FOR CLARIFICATION

Originator: D.A.ROBINSON Date: OCTOBER 27, 1992

Technical Clarification Title: Service Water System sampling when PCCW Rad. Monitor(s) is (are) out of service.

Technical Clarification No.: TS-174

Type of Clarification:

Tech Spec [X] (TS) FSAR(Excluding 17.2) [ ] (FS) FSAR 17.2 [ ] (QS) Licensing [ ] (LS)

REQUEST FOR CLARIFICATION: (Attempt to state the request as a question.)

Tech. Spec. 3.3.3.1, 3.3.3.9, and ODCM PART A, Section 3.0, Table A3.1 Item D, Note 7 require sampling of the PCCW and SW systems for gamma activity for leak detection when the Radiation Monitor(s) (1-RM-6515 or 1-RM-6516) is (are) Inoperable.

When PCCW is shutdown and or drained the PCCW System is monitored utilizing the guidance in Technical Clarification No. TS-83.

When Service Water is drained a sample from the PCCW/SW Heat Exchanger effluent is unavailable. What action is required when the Service Water side of the PCCW/SW Heat Exchanger is drained.

CONCURRENCE: W.B. [signature] 10/28/92
Group Manager (Date)

SECTION II - INITIATION

RECEIVED REGULATORY COMPLIANCE: [signature] 11/13/92
Lead Engineer - Compliance (Date)

GROUP MANAGER ASSIGNED: J.M. Peschel
(Name/Dept) (Date)

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SECTION III - EVALUATION

The following actions are required when the Service Water side of the Primary Component Cooling Water (PCCW) Heat Exchanger is drained and grab samples of the Service Water System are required:

a. Grab samples from the Service Water System will be obtained at the frequencies specified in Technical Specification 3.3.3.1, 3.3.3.9, and the Offsite Dose Calculation Manual as the Service Water System is being drained until obtaining these samples is not physically possible.

b. Grab samples are not required once the Service Water System is drained such that it is not physically possible to obtain the samples.

c. When refilling the Service Water System, grab samples shall resume as soon as physically possible, at the intervals specified in the aforementioned sources, and continue until the PCCW radiation monitors (1-RM-6515 and 1-RM-6516) are OPERABLE.

Sampling of the PCCW system with the Service Water system drained and the PCCW system in operation shall continue per the requirements of Technical Specifications 3.3.3.1 and 3.3.3.9 and the guidance of Technical Clarification TS-083.

Prepared By: [signature] 12/1/92 (Date)
Concurrence: [signature] 12/1/92 Cognizant Group Manager (Date)

SECTION IV - REVIEW AND APPROVAL

(Check Appropriate Boxes)

[] [signature] 12/1/92 Regulatory Compliance Manager (Date)	[] ______ (Date)
[] [signature] 12-7-92 Station Manager (Date)	[] ______ (Date)
[] ______ (Date)	[] ______ [] ______ (Date)

SORC MEETING NO.: 92-220 DATE: 12-7-92

APPROVED BY:

[signature] 2/2/93
Executive Director - Nuclear Production (TS) (Date)

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Executive Director - Engineering and Licensing (FS) (Date)

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Director of Licensing Services (LS) (Date)

______
Director of Quality Programs (QS) (Date)

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TS-174

BACKGROUND INFORMATION FOR TS-174

The purpose of the plant radiation monitors is to sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. In the case of the Primary Component Cooling Water (PCCW) loops, the radiation monitors (1-RM-6515 and 1-RM-6516) sense radiation in the PCCW system which could leak into the Service Water System and be discharged to the environment via the multiport diffuser. Per Technical Specification 3.11.1.1, the concentration of radioactive material released in liquid effluents at the point of discharge from the multiport diffuser must be within specified limits. This limitation provides assurance that the levels of radioactive materials in unrestricted areas will not pose a threat to the health and safety of the public.

Based on the importance of maintaining radioactive effluent releases within limits that guarantee the health and safety of the public will not be at risk, the PCCW radiation monitors are required to be in operation at all times. When a radiation monitor is inoperable, grab samples from the PCCW and Service Water systems must be obtained and analyzed as a compensatory measure in accordance with Technical Specification 3.3.3.1, Table 3.3-6 Action 28, Technical Specification 3.3.3.9, Table 3.3-12, Action 32, and Part A, section 3.0 of the Offsite Dose Calculation Manual. If the service water system is drained, there is no potential for inadvertent radioactive liquid effluent release through the service water system to the environment via the multiport diffuser. Thus, when the system is drained there is no need to obtain the grab sample. However, when the system is being filled, grab samples must be obtained as soon as possible to ensure that the water discharged to the environment is in compliance with Technical Specification 3.11.1.1.

The purpose of the PCCW monitors is to detect radioactivity indicative of a leak from the Reactor Coolant System or from one of the other radioactive systems which exchange with the PCCW System. These monitors are required to be operable at all times. Grab samples of PCCW are required when the PCCW monitors are not operable. Since the purpose of obtaining the PCCW samples is to provide an indication of a leak of radioactive liquid into the PCCW system, draining of the Service Water system does not remove the reason for obtaining the PCCW grab samples. These samples shall be obtained as specified in Technical Specifications 3.3.3.1 and 3.3.3.9. See Technical Clarification TS-083 for further guidance concerning PCCW grab samples.

This determination is consistent with the Bases for Technical Specifications 3.3.3.1, 3.3.3.9, and 3.11.1.1.

39r

Question Number:	SRO 40
Question:	Plant conditions: • A small break LOCA has occurred. • The crew is currently performing the actions of ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. • All critical safety functions are satisfied with the exception of containment (Z) which has a YELLOW terminus based on post accident monitor radiation >10 Rem/hr. • With an RCS cooldown in progress the Unit Supervisor refers to FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL. What mitigating actions are directed by this procedure? A. FR-Z.3 directs that the Containment Online Purge (COP) system be placed in service to cleanup the containment atmosphere. B. FR-Z.3 directs performance of a containment bleed through the Combustible Gas Control System to the Enclosure Air Handling Filter units. C. FR-Z.3 directs that the Containment Recirculation Filter System be placed in service in the "Filter" mode. D. FR-Z.3 directs that the Containment Air Purge (CAP) system be placed in service in the refueling purge mode.
Answer:	C
Justification:	C is correct as per step #2 of FR-Z.3. A & D are incorrect as step #1 of FR-Z.1 check to ensure that COP & CAP containment isolation valves are isolated. B is incorrect as this action would only be directed by the TSC as a possible course of action to reduce containment hydrogen concentration.
Direct/New/ Modified	New
K/A #:	E16EK1.2
K/A Values:	3.2
Cognitive Level:	Application (III)
References:	FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL

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Question Number:	SRO 41
Question:	The plant is in MODE 6. You are assigned as the Refueling SRO in containment. • Fuel moves are in progress. • There is a spent fuel assembly in the refueling machine mast. — near the core • Refueling Cavity level has been DECREASING at approximately 0.5 inches per minute due to a failed RHR suction relief valve. • Refueling Cavity level is currently 17 feet above the reactor vessel flange and DECREASING. Per procedure, Where do you direct the refueling machine operator to place the spent fuel assembly? IAW A. In any core location B. In the RCCA change fixture C. In the upender in a vertical position D. In the transfer canal with the refueling machine mast fully extended
Answer:	A
Justification:	A is correct because it is the direction provided in the reference. B & C are incorrect because the fuel assembly may be exposed if level continued to decrease. D would only be correct if the fuel assembly could not be moved to a core location.
Direct/New/Modified	Modified from 1998 NRC exam
K/A #:	000036AA1.04 2.2.29
K/A Values:	3.1/3.7 1.6/3.8
Cognitive Level:	Memory (I)
References:	OS1215.05, Loss of Refueling cavity water Objective L1192I02RO

Question Number:	SRO 42
Question:	Plant conditions: • The plant has tripped due to a spurious closure of the MSIVs. • The crew has transitioned from E-0, REACTOR TRIP OR SAFETY INJECTION, to ES-0.1, REACTOR TRIP RESPONSE. • 'A', 'B', & 'C' Steam Generator pressures are approximately 1125 psig with pressure being controlled by their respective ASDVs. • The ASDV for the 'D' S/G has failed and cannot be controlled from the Control Room. • The "S/G Safety Valve Open" alarm is in on VA3 and 'D' S/G pressure indicates approximately 1190 psig. • The Shift Manager announces that all critical safety functions are satisfied with the exception of Heat Sink (H) which is YELLOW due to the elevated pressure in 'D' Steam Generator. What procedure should be used to deal with this condition? required A. ES-0.1, REACTOR TRIP RESPONSE B. FR-H.4, RESPONSE TO LOSS OF NORMAL STEAM DUMP CAPABILITIES C. FR-H.2, RESPONSE TO STEAM GENERATOR OVERPRESSURE D. E-2, FAULTED STEAM GENERATOR ISOLATION
Answer:	B
Justification:	B is correct, the H terminus would be YELLOW directing the crew to FR-H.4 under these conditions. A is incorrect, while the RNO to step #8 of ES-0.1 directs the use of the ASDVs if Steam Dumps are not available, no contingency actions are provided for local ASDV operation or opening of the MSIV bypass valves if necessary as in FR-H.4. C is incorrect, FR-H.2 is only used if S/G pressures are greater than highest setpoint safety valve (1255 psig). D is incorrect, no conditions in the stem indicate a faulted S/G and a transition from ES-0.1 to E-2 cannot be made directly.
Direct/New/ Modified	New
K/A #:	E13EA2.1
K/A Values:	2.9/3.4
Cognitive Level:	Analysis (III)
References:	F-0.3, HEAT SINK CSF STATUS TREE. FR-H.4, RESPONSE TO LOSS OF NORMAL STEAM DUMP CAPABILITIES Objective L1211I04RO

minimum CNMT Level for

Too generic

Question Number:	SRO 43
Question:	Which of the following identifies a containment flooding condition? A. Containment building level indicates greater than 5 feet. B. Containment Sump 'A' off-scale high. C. Containment Sump 'B' off-scale high. D. Containment Building level indicates greater than 2.5 feet.
Answer:	A
Justification:	A is correct in accordance with F-0.5, "Containment CSF status Tree," an orange condition is generated when Containment Building level is not less than 5 feet. B & C are incorrect as these two sumps fill before Containment Building level comes on-scale. D is incorrect based on the correct answer.
Direct/New/ Modified	New
K/A #:	E15EK1.3
K/A Values:	2.8/3.0
Cognitive Level:	Memory (I)
References:	F-0.5, Containment (2) CSF Status Tree Objective L1212I09RO

All can be possible answers as written

Question Number:	SRO 44
Question:	During a reactor startup the Primary Board Operator is withdrawing shutdown Bank E. Which of the following represents the speed at which the shutdown rods should be moving? A. 32 Steps Per Minute B. 48 Steps Per Minute C. 64 Steps Per Minute D. 72 Steps Per Minute
Answer:	C
Justification:	Manual control of shutdown banks is accomplished at 64 SPM. A is incorrect because no rod group moves at 32 SPM when in MANUAL. A is plausible because the rod speed for shutdown bank E is not displayed on the MCB. B is incorrect because it represents the speed of the control banks in MANUAL. D is the fastest that the control banks will move in AUTO.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	001K4.14
K/A Values:	2.6/2.8
Cognitive Level:	Memory (I)
References:	Lesson Plan L1113I, Section 3.2.3, Objective L1113I04RO

Question Number:	SRO 45
Question:	Plant conditions: • 90% power. • A rapid power decrease is in progress due to increasing vibration levels on the 'A' RCP. • RCP frame vibration ~~spikes to 6~~ mils on MPCS color graphics and the BOP operator confirms ~~this vibration~~ level using hardwire instrumentation on MCB-GR. required 8 What action should ~~be taken in accordance~~ with OS1201.01, RCP MALFUNCTION? A. Continue the rapid power decrease to drive power below P-8 then trip the RCP while manually controlling 'A' S/G level. B. Place 'A' S/G main feed regulating valves to manual and feed the generator to 60% to 70% NR level. Trip the 'A' RCP and shutdown to MODE 3. C. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION, and immediately trip the 'A' RCP. D. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION, step 1. Trip the 'A' RCP after the immediate actions are complete.
Answer:	D
Justification:	A is incorrect, an immediate Rx trip is required per the abnormal. B is incorrect because an immediate reactor trip is required, the actions listed would only be taken if reactor power were below P-8. C is incorrect because actions to trip the affected RCP are not taken until after the immediate actions are complete. D is correct per the RNO to step #2 for actions with frame vibration greater than danger value of 5 mils and reactor power greater than P-8 (50% power)
Direct/New/ Modified	New
K/A #:	003A2.02
K/A Values:	3.7/3.9
Cognitive Level:	Memory (I)
References:	OS1201.01, RCP MALFUNCTION Objective L1181I03RO

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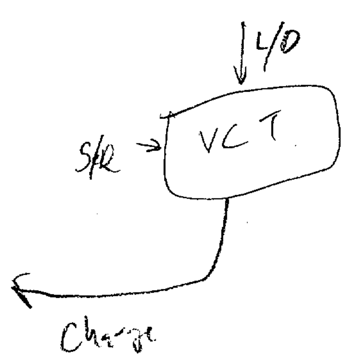
Question Number:	SRO 46 spell out
Question:	The operator is using MCB indications to perform a CVCS flow balance. Assuming RCS temperature and pressurizer level are stable, which of the following describes the CVCS system flow balance? A. Letdown + seal return = indicated charging flow + seal injection flow B. Letdown + seal return = indicated charging flow C. Letdown – seal injection flow = indicated charging flow D. Letdown – seal injection flow = indicated charging flow + seal return flow
Answer:	B
Justification:	The flow balance is charging flow (which includes seal injection) should equal letdown flow plus seal return flow. (B correct)
Direct/New/ Modified	Direct from bank
K/A #:	004K4.05
K/A Values:	3.3/3.2
Cognitive Level:	Analysis (III)
References:	P&ID CS-B20725, CVCS text Objective L1105I02RO



Seabrook SRO Examination
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Question Number:	SRO 47
Question:	Plant conditions: • A reactor trip and safety injection has occurred. • All Steam Generator Pressures are DECREASING • Containment temperature, pressure, and humidity are INCREASING • Tave is DECREASING • Containment Pressure is currently NON-adverse Bold For this event, which of the following actuations is designed to prevent the containment from exceeding its design pressure limit? A. Containment Isolation Phase B B. Main Steam Line Isolation C. Containment Isolation Phase A D. Feedwater Isolation
Answer:	B.
Justification:	A is incorrect because Phase B Isolation is designed to isolate remaining containment penetrations not isolated by Phase A. B is correct IAW the reference and the design basis of the MSIS. C is incorrect because Phase A is designed to isolate non-essential process lines that do NOT increase the potential for damage to equipment inside containment. D is incorrect because Feedwater isolation is designed to minimize excessive cooldown and protect the turbine from moisture carryover
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	013A1.02
K/A Values:	3.9/4.2
Cognitive Level:	Analysis (III)
References:	IS system text, sections 3.3.3, 3.3.3.1, 3.3.3.3, and 3.3.4 Lesson Plan L1139I, Objective L1139I08RO

Question Number:	SRO 48
Question:	The plant is at 88 % power. Control Bank D Group Demand Counters indicate 228 steps. Due to an Urgent Failure of DRPI Data B, the Accuracy Mode Selector Switch is placed in the DATA A position. How does this affect the OPERABILITY of Rod Position Indication? A. Rod Position Indication is INOPERABLE. THERMAL POWER must be reduced to less than 50% of RATED THERMAL POWER within 8 hours. B. Rod Position Indication is OPERABLE and capable of determining rod position within ± 12 steps. C. Rod Position Indication is INOPERABLE. POWER OPERATION may continue as long as the affected rod positions are determined indirectly by the Incore Detector System within 8 hours. D. Rod Position indication is OPERABLE and capable of determining rod position within ± 6 steps.
Answer:	B.
Justification:	A is incorrect. Action stated is for inop DRPI. B is correct IAW the listed reference, -10/+4 steps C is incorrect. The action stated is for inoperable DRPI D is incorrect, DRPI accuracy is +10/-4 steps
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	014K5.02
K/A Values:	2.8/3.3
Cognitive Level:	Application (III)
References:	RPI system text, figure RPI-3.2 Lesson Plan L1114I, Objective L1114I08RO

Question Number:	SRO 49
Question:	The plant is at 7 % power during a plant startup. Intermediate Range channel N36 fails HIGH. Which of the following describes the effect of this instrument failure? A. The startup may continue after bypassing C-1 for IR N35. B. The startup may continue after bypassing C-1 for IR N35 and IR N36. C. The reactor must be placed in HOT STANDBY within 6 hours. P-6 will not automatically energize with IR N36 failed high. D. The reactor will trip on High Intermediate Range Flux.
Answer:	D.
Justification:	A and B are incorrect because the high IR flux trips (same switch as C-1 bypass) cannot be bypassed below P-10 (10% power) C is incorrect because although the distracter is a standard requirement for equipment failure, in this case an automatic trip will occur D is correct because a reactor trip will occur with 1 out of 2 IR channels indicating 25% equivalent reactor power
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	015K1.01
K/A Values:	4.1/4.2
Cognitive Level:	Comprehension (II)
References:	RP system text, section 4.2.2 NIS Lesson Plan L1112I, Objective L1112I08RO

Question Number:	SRO 50
Question:	Which of the following instruments provide input to Train 'B' of the RCS Subcooling Monitor? A. RCS Wide Range Pressure instrument PT-403 and the average of all core exit thermocouples B. RCS Wide Range Pressure instrument PT-405 and the auctioneered high core exit thermocouple C. RCS Wide Range Pressure instrument PT-403 and the auctioneered high average quadrant temperature D. RCS Wide Range Pressure instrument PT-405 and the auctioneered high average quadrant temperature
Answer:	C
Justification:	A is incorrect because there is no input from an average of "all" CET's. B is incorrect because the subcooling monitor utilizes auctioneered high average quadrant temperature. C is correct according to lesson plan L1140I. D is incorrect because PT-405 inputs to ICCM A.
Direct/New/ Modified	Modified from 1998 NRC exam.
K/A #:	017K4.01
K/A Values:	3.4/3.7
Cognitive Level:	Memory (I)
References:	Lesson Plan L1140I Objective L1140I013RO

Question Number:	SRO 51
Question:	Plant conditions: • A LOCA is in progress. • Containment pressure currently indicates 16 PSIG and decreasing slowly. • Both trains of CBS are operating. • The Containment pressure recorders indicate that pressure increased to a peak of 21 psig. • All Containment Phase B penetrations are isolated and no safeguards actuation signals have been reset. Which of the following indicates the expected status of Containment cooling systems? A. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the RECIRC MODE. B. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the FILTER MODE. C. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the FILTER MODE. D. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the RECIRC MODE.
Answer:	D.
Justification:	A 'P' signal (Hi-3, 18 psig) will trip Containment Structure Cooling fans and CRDM cooling fans, and start Containment Recirc Fans in the RECIRC Mode. FILTER Mode is used for containment pre-entry. A 'P' signal deenergizes the solenoids in the dampers to cause the system to fail to the recirc MODE
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	022A3.01
K/A Values:	4.1/4.3
Cognitive Level:	Analysis (III)
References:	CHV system text, sections 4.1.3 and 4.1.4 Lesson Plan L1120I, Objective L1120I04RO

Question Number:	SRO 52
Question:	The plant has sustained a Large Break LOCA. The following conditions exist: • Cold Leg ECCS Flow on SI-FI-917 indicates 900 gpm • Safety Injection flow is 600 GPM in EACH train • RHR flow is 3700 GPM in EACH train • Train A CBS pump is running with discharge pressure at 190 psig • Train B CBS pump did NOT start upon actuation of CBS Assuming the RWST was at it's Tech Spec minimum level when the event occurred, approximately how much time will pass before initiation of swapover to Cold Leg recirculation? A. 15 minutes B. 30 minutes C. 45 minutes D. 60 minutes
Answer:	B.
Justification:	TS minimum RWST level is 477K gallons. Swapover occurs at approximately 120,000 gallons. Approximately 350-355K gallons will be pumped into containment at the following total flow rate: Charging, 900 GPM; SI, 1200 GPM; RHR, 7400 GPM; CBS, 3000 GPM. Total flow = 12,500 GPM. 350K gl/12.5K GPM = 28 minutes
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	026K3.02
K/A Values:	4.2/4.3
Cognitive Level:	Synthesis (III)
References:	CBS system text, section 4.1.3 Lesson Plan L1117I, Objectives L1117I10, L1117I13, and L1117I14

~~CS flow rate~~

Question Number:	SRO 53
Question:	The plant is at 96 % power. All control systems are aligned for AUTOMATIC operation. A 4.16KV bus 3 fault causes the incoming UAT feeder to bus 3 to trip open. With NO operator action, which of the following describes the response of the plant? A. The reactor will trip on LO-LO SG levels. B. The alternate supply breaker to bus 3 from the RAT will close. C. DG 'A' and 'B' will automatically start and supply busses E5 and E6. D. The reactor will trip on Loss of RCS flow.
Answer:	A.
Justification:	Bus 3 supplies 2 condensate pumps and 1 heater drain pump. At 97% power, a loss of these pumps will cause a low suction pressure trip of the main feed pumps, resulting in Lo-Lo SG levels. B is incorrect because the RAT will not close in with a UAT lockout. C is incorrect because Bus E6 is unaffected. D is incorrect because RCP's are 13.8 KV bus 1 or 2 loads.
Direct/New/ Modified	Direct from 1998 NRC exam **NOTE: A minor change was made to the stem after validation. The change clarified the stem and made it more technically correct.**
K/A #:	056A2.04
K/A Values:	2.6/2.8
Cognitive Level:	Application (III)
References:	4.16KV text, section 4.1.1.3 Lesson Plan L1093I, Objectives L1093I05,06,07

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 54
Question:	OS1235.03, SG LEVEL INSTRUMENT FAILURE, contains the following CAUTION statement prior to step #1: "During operation in manual feedwater control at ≥ 65% power, maintain Steam Generator water level 50% to 70% narrow range." What is the basis for this CAUTION? A. Limits the mass in the SGs with respect to the UFSAR steam break analysis. B. Limits the mass in the SGs in consideration of SG overfill during a Steam Generator Tube Rupture Event. C. Provides adequate mass to ensure iodine partitioning during a Steam Generator Tube Rupture Event. D. Provides adequate mass to maintain heat sink during loss of all AC power.
Answer:	C
Justification:	C is correct in accordance with lesson L1193I. A is incorrect as NR level up to 70% is in excess of that assumed in the steam break analysis. B is incorrect as this action maximizes, not limits, S/G water level. D is incorrect, heat sink is not an initial concern in a loss of all AC power event.
Direct/New/ Modified	Modified from bank
K/A #:	2.4.11
K/A Values:	3.4/3.6
Cognitive Level:	Analysis (III)
References:	OS1235.03, SG LEVEL INSTRUMENT FAILURE, lesson plan Objective L1193I13RO

Memory?

Question Number:	SRO 55
Question:	Plant conditions: • A reactor trip has occurred due to a loss of offsite power. Safety Injection has actuated. • The Emergency Diesels are powering their respective emergency 4160 V buses. • Neither EFW pump is running • The crew is carrying out the actions of FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, and has aligned the SUFP to bus E5. • The BOP operator attempted to start the SUFP but the amber "Breaker Disagreement" light energized when the pump control switch was taken to START. What caused the pump ~~start failure~~? Spell out A. The reactor trip breakers must be cycled before the pump will start. B. RMO on bus E5 must be reset before the pump will start. C. The Safety Injection signal must be reset before the pump will start. D. The UAT or RAT breaker to bus E5 must be closed (EPS reset) before the pump will start.
Answer:	B
Justification:	A is incorrect, P-4 does not input to the SUFP start circuit. B is correct per schematic 1-NHY-310844 shA93b. EPS RMO must be reset after the EPS has completed sequencing. C is incorrect, an SI signal only blocks auto start of the SUFP, it will not prevent a manual start. D is incorrect, while this would allow start of the SUFP it is NOT a condition that must exist. As the pump will start on bus E-5 when powered by the EDG when RMO is reset.
Direct/New/ Modified	New
K/A #:	0612K2.02
K/A Values:	3.7/3.7
Cognitive Level:	Comprehension (II)
References:	1-NHY-310844, sh. A93b & CN1a

B & D

Backward Logic?

Frequently Occurs in EOPs

Operational Valid

Question Number:	SRO 56
Question:	Plant conditions: • A loss of offsite power has occurred. • The 'B' Emergency Diesel Generator has failed to start. • The 'A' Emergency Diesel Generator is powering bus E5. Which of the following describes the expected electrical power flowpath to PP-1A, PP-1B, and PP-1F? A. Bus E51→MCC E512→UPS-I-1A→PP-1A Battery B-1B→DC bus 11B→UPS-I-1B→PP-1B Battery B-1B→DC bus 11B→UPS-I-1F→PP-1F ? Distractors are unclear to examiner B. Battery Charger BC-1A→DC bus 11A→UPS-I-1A→PP-1A Bus E61→MCC E612→UPS-I-1B→PP-1B Bus E61→MCC E612→UPS-I-1F→PP-1F C. Battery B-1A→DC bus 11A→UPS-I-1A→PP-1A Battery Charger BC-1B→DC bus 11B→UPS-I-1B→PP-1B Battery B-1B→DC bus 11B→UPS-I-1F→PP-1F D. Battery Charger BC-1A→DC bus 11A→UPS-I-1A→PP-1A Battery B-1B→DC bus 11B→UPS-I-1B→PP-1B Bus E63→MCC E631→480/120v transformer via Static Transfer switch→PP-1F
Answer:	A
Justification:	A is correct. B is incorrect because UPS-I-1A would not be supplied with DC power under these conditions and PP-1B & 1F would not be supplied with AC power. C is incorrect because PP-1A would not be fed from DC power and no AC power is available to power BC-1B. D is incorrect because power would not be available to alternate AC power supply from Bus E63 and UPS-I-1A would not be supplied with DC power under these conditions
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	063K1.03
K/A Values:	2.9/3.5
Cognitive Level:	Memory (I)
References:	125 VAC Electrical System Detailed System Text Objective L1098I03RO

Question Number:	SRO 57
Question:	Which of the following radiation monitors is both a release path monitor AND has an automatic isolation function associated with it? A. 1GM810, Condenser Air Evacuation. B. 1LM216, SG 'A' Blowdown Line Monitor C. 1LM241, High Range Letdown Activity D. 1LM220, PCCW Loop A
Answer:	B
Justification:	A, C, & D are incorrect because although they are release path monitors, they have no automatic functions associated with them. B is correct as 1LM216 will close SB Flash Tank Outlet Valve SB-CV-6519.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	068A4.04
K/A Values:	3.8/3.7
Cognitive Level:	Memory (I)
References:	OS1252.01, PROCESS OR EFFLUENT HIGH RADIATION, Attachment A Objective L1141I06RO

Question Number:	SRO 58
Question:	Plant conditions: • The Waste Gas system is operating in the "Purge" mode of operation with gas being released to the plant vent via PAH-F-16. • Processing of gas from a recent failed fuel event causes a high radiation condition. • All radiation monitors in the system are in high alarm. What automatic actions are expected to occur under these conditions? A. RM-6503, Waste Gas Compressor Inlet Monitor, automatically closes WG-PCV-1491 isolating the 100 psig header. B. RM-6502, Carbon Delay Bed Inlet Monitor, automatically closes VG-PCV-1713 isolating the inlet to the Carbon Delay Beds. C. RM-6502, Carbon Delay Bed Inlet Monitor, automatically closes VG-V50 terminating purge release to PAH-F-16. D. RM-6504 automatically closes WG-FV-1602 terminating the purge release to PAH-F-16.
Answer:	D
Justification:	A, B, & C are incorrect, RM-6502 & 6503 have no process isolation functions. D is correct, High alarm on RM-6504 closes WG-FV-1602 isolating purge release to the plant vent via PAH-F-16.
Direct/New/ Modified	New
K/A #:	017A3.03
K/A Values:	3.7/3.8
Cognitive Level:	Memory (I)
References:	OS1252.01, Attachment A Lesson 1146, Waste Gas Systems, L1146I07RO

Question Number:	SRO 59
Question:	Which of the following plant AREA radiation monitors provides an automatic actuation function? A. RM 6518, High Range Spent Fuel Pool B. RM 6540, PAB Volume Control Tank Area C. RM 6535A, Manipulator Crane Train A D. RM 6576A, Containment Post LOCA Train A
Answer:	C
Justification:	A, B, & D are incorrect, these area monitors provide no automatic actuation functions. C is correct, RM 6535A in high alarm generates a containment ventilation isolation signal which closes the CAP and COP containment isolation valves.
Direct/New/ Modified	New
K/A #:	072K4.03
K/A Values:	3.2/3.6
Cognitive Level:	Memory (I)
References:	OS1252.03, AREA HIGH RADIATION Lesson 1120, CHV, L1120I23RO

Q Points to B

Question Number:	SRO 60
Question:	Plant conditions: • Plant was at 100% power with all control systems in AUTOMATIC when the 'B' MFP tripped, inducing a turbine setback. • The unit supervisor has entered OS1231.03, TURBINE RUNBACK/SETBACK. • D7762 "CNTL BK D INSERTION LIMIT LO-LO" alarms on VA2 What is the significance of this alarm? Fixed Incore Detectors ? A. Axial flux difference has entered the "FID Dependent" space. B. The MODE 1 Shutdown Margin Limit may have been exceeded. C. The setback rate has exceeded the capabilities of the Control Rod Drive System. D. A malfunction has occurred in the Rod Control System.
Answer:	B
Justification:	A is incorrect because AFD limits are not necessarily exceeded if the rods are inserted below the RIL. C is incorrect because the alarm setpoint for RIL LO-LO is independent of any rate of turbine load decrease (i.e. if the rods were able to insert faster to keep Tavg on program the RIL would still be exceeded). D is incorrect because this alarm does not indicate a rod control system problem.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	001K5.04
K/A Values:	4.3/4.7
Cognitive Level:	Comprehension (II)
References:	Alarm response procedure for D7762, CNTL BK D INSERTION LIMIT LO-LO OS1202.04, RAPID BORATION, Entry Conditions Objective L1183I09RO

Yes or no A+B?

Will Review

Rewritten - New Q

Question Number:	SRO 61
Question:	Plant conditions: • All Containment Air Handling units are in service with the exception of CAH-FN-1F which is in Pull-to-Lock. • A loss of offsite power causes a plant trip. • The emergency diesel generators re-power their respective emergency buses. • After the EPS loading sequence is complete a safety injection (SI) actuation occurs. What is the present condition of the Containment Air Handling units as a result of these events? A. All of the CAH fans that were previously running were restarted at step #3 of the EPS sequence. B. All of the CAH fans that were previously running were restarted by the Safety Injection actuation. C. To prevent an overload condition on the emergency buses the EPS started only CAH-FN-1A and CAH-FN-1C. D. None of the CAH fans are running.
Answer:	D
Justification:	C is incorrect, the previously running fans were restarted by the EPS at step 3 in response to the LOP. A is incorrect, the safety injection (SI) actuation opens contacts in the start CKT for each fan to prevent an auto start. D is correct, B & C are incorrect, the SI also energizes the RA relay in the EPS which energizes the trip coil for each fan breaker preventing an auto start.
Direct/New/ Modified	New
K/A #:	022A4.01
K/A Values:	3.6/3.6
Cognitive Level:	Comprehension (II)
References:	Schematic diagram 1-NHY-310931 Lesson Plan 1120, objective L1120I04RO

Question Number:	SRO 62
Question:	Plant conditions: • Stable at 100% power • D5816-"Non-Vital Inst. Panel 5 Power Lost" annunciates on VA4. What affect does a loss of power to ED-PP-5 have on the plant? A. Loss of automatic operation of both PZR PORVs. B. Loss of safeguards equipment actuation on an ESFAS signal. C. Main feed pump speed control signals are lost and the pump recirculation valves fail open. D. All main feed regulating valves fail closed.
Answer:	C
Justification:	C is correct in accordance with ON1247.03. B & D are incorrect, these failures result form loss of power to vital DC bus 11A or 11B. A is incorrect, loss of power to PP-5 affects only the 'A' train PORV. Operation of both PORVs is affected by loss of power to PP-1C or PP-1D.
Direct/New/ Modified	New
K/A #:	059K6.09
K/A Values:	2.6/2.6
Cognitive Level:	Memory (I)
References:	OS1247.03, LOSS OF 120VAC NON-VITAL PANEL PP-5 Objective L1186I02RO

Question Number:	SRO 63
Question:	The plant has sustained a Small Break LOCA. The following conditions exist: • PORV 456B is stuck OPEN, and has NOT been isolated • RCS pressure is 1050 psig • Core Exit Thermocouples are approximately 550°F • All RCPs are TRIPPED. Which of the following instruments will provide the most reliable indication of actual RCS inventory? A. Pressurizer Hot-calibrated level instrument LT-459 B. Pressurizer Cold-calibrated level instrument LT-462 C. Reactor Vessel Dynamic Range Level (RVLIS) D. Reactor Vessel Full Range Level (RVLIS)
Answer:	D.
Justification:	A and B are incorrect because pressurizer level is not an accurate indication of inventory with a hole in the pressurizer. C is incorrect because it is most reliable when RCP's are running. D is correct because it measures DP across the vessel under static conditions.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	002K1.07
K/A Values:	3.5/3.7
Cognitive Level:	Memory (I)
References:	Lesson Plan L1140I, Objective L1140I08RO AM System text section 3.2.3

Question Number:	SRO 64
Question:	A Large Break LOCA has occurred. All safeguards equipment functioned as designed. RWST LO-LO level alarm is actuated. How will swapover to Cold Leg recirculation be performed? A. Containment recirculation sump valves, CBS-V8 and CBS-V14, will automatically open. RWST suction valves, CBS-V2 and CBS-V5, will automatically close when the containment recirculation suction valves are fully open. B. Containment recirculation sump valves, CBS-V8 and CBS-V14, will automatically open. RWST suction valves, CBS-V2 and CBS-V5, must be manually closed when the containment recirculation valves are open. C. Containment recirculation sump valves, CBS-V8 and CBS-V14, must be manually opened. RWST suction valves, CBS-V2 and CBS-V5, must be manually closed. D. Containment recirculation sump valves, CBS-V8 and CBS-V14, must be manually opened. RWST suction valves, CBS-V2 and CBS-V5, automatically close when the containment recirculation valves are open.
Answer:	B.
Justification:	As long as an S signal is present, the containment sump suction valves, CBS-V8 and CBS-V14, will auto open. After they are open, S can be reset, and the RWST suctions, CBS-V2 and CBS-V5, may be manually closed from the control room.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	006A4.05
K/A Values:	3.9/3.8
Cognitive Level:	Memory (I)
References:	CBS system text, figure 3.2, section 4.1.3 Lesson Plan L1117I, Objective L1117I13RO

Question Number:	SRO 65
Question:	Plant conditions: • Reactor power is 75% and a power increase to 100% is in progress. • All control systems are in AUTOMATIC. • Backup heater group 'A' is selected to ON to force PZR spray. All other heater groups are in AUTO. • Pressurizer pressure is 2235 psig and stable. • A failure in the master pressure controller circuitry causes the controller setpoint to drift to 2300 psig. • The failure has not yet been diagnosed by the crew. Which of the following describes the INITIAL control system response to this condition? A. PZR spray valves CLOSE and all PZR heater groups ENERGIZE. B. PZR spray valves OPEN and all PZR heater groups except 'A' DE-ENERGIZE. C. PZR spray valves CLOSE and all PZR heater groups except 'A' DE-ENERGIZE. D. PZR spray valves OPEN and all PZR heater groups ENERGIZE.
Answer:	A
Justification:	A is correct, an increase in master controller setpoint creates a deviation between set pressure (2300 psig) and actual pressure (2235 psig). The control system will attempt to increase pressure towards setpoint by closing the spray valves and energizing all PZR heaters. B & D are incorrect, they state that spray valves will open. C is incorrect, it states that all heater groups (with the exception of 'A') will de-energize.
Direct/New/ Modified	New
K/A #:	010A3.02
K/A Values:	3.6/3.5
Cognitive Level:	Memory (I)
References:	Lesson Plan, 1108, Objectives L1108I10RO & L1108I08RO

Question Number:	SRO 66
Question:	Plant conditions: • MODE 2, 2% power, plant startup in progress • Pressurizer pressure channel PT-455 has failed low. • The crew has tripped all required bistables in accordance with OS1201.06, PRESSURIZER PRESSURE INSTRUMENT PT 455/458 FAILURE. • Power is subsequently lost to PP-1B. Which of the following describes the expected plant response to these events? A. The plant trips due to LOW PRZ pressure or OPΔT, ONLY 'A' train SI actuates. B. The plant trips due to HIGH PZR pressure or OTΔT, BOTH trains of SI actuate. C. The plant trips due to LOW PZR pressure or OPΔT, BOTH trains of SI actuate. D. The plant trips due to HIGH PZR pressure or OTΔT, ONLY 'A' train SI actuates.
Answer:	D
Justification:	D is correct, the reactor will trip on redundant bistable actuation for PZR pressure high or OTΔT. A & C are incorrect, The reactor will not trip on low PZR pressure as this trip is blocked below 10% power. PZR pressure does not input into the OPΔT trip. B & C are incorrect, Loss of power to PP-1B will prevent auto actuation of 'B' train SI.
Direct/New/ Modified	New
K/A #:	012K1.01
K/A Values:	3.4/3.7
Cognitive Level:	Analysis (III)
References:	OS1247.01, LOSS OF 120 VAC VITAL INST PANEL PP-1A, 1B, 1C, OR 1D Objective L1186I09RO

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Question Number:	SRO 67
Question:	Plant conditions: • MODE 1, 60% power. • All control systems are in AUTOMATIC • The controlling NR level channel on the 'A' SG has failed low • The Unit Supervisor has entered OS1235.03, SG LEVEL INSTRUMENT FAILURE • The 'A' main feed regulating valve (MFRV) controller will not shift into MANUAL. What actions are directed by the procedure for this event? A. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. B. Immediately swap the 'A' SG controlling NR level channel to an alternate channel. C. Control feedwater flow using main feed pump speed controllers while taking local control of the affected MFRV. D. Control feedwater flow to the 'A' SG using the MFRV bypass valve.
Answer:	C
Justification:	C is correct, OS1235.03, step 1b RNO directs "Control feedwater flow using main feed pump speed controllers. Refer to OS1090.01, MANUAL OPERATION OF REMOTE OPERATED VALVES and locally control affected SG feed reg valve." A is incorrect, a reactor trip is not directed by this procedure. B is incorrect, while this may correct the problem, this action is not directed until step #2 after the affected MFRV is in manual control. D is incorrect, this failure causes the affected MFRV to open. The by-pass valve will do nothing for feed control under these conditions.
Direct/New/ Modified	New
K/A #:	016K3.12
K/A Values:	3.4/3.6
Cognitive Level:	Memory (I)
References:	OS1235.03, step 1b RNO, Objective L1193I03RO

are both plausible
Requires detailed knowledge of procedure

Question Number:	SRO 68
Question:	Plant conditions: • A large break LOCA has occurred • The crew is performing the actions of E-1, LOSS OF REACTOR OR SECONDARY COOLANT. • Containment pressure peaked at 25 psig and is currently 16 psig and DECREASING slowly. • All safeguards equipment is functioning as designed. • Only SI has been RESET. • A loss of off-site power occurs. Spell out Which of the following describes the expected response of the containment structure recirc. filter fans (FN-3A & 3B)? A. Neither fan re-starts because containment pressure is below 18 psig. B. Both fans ~~re-start~~ when the Diesel Generator Breakers close onto buses E5 and E6 (step 0). ? C. Neither fan re-starts because the SI signal is reset. ? D. Both fans re-start when the EPS sequence is complete (Step 9).
Answer:	B
Justification:	A is incorrect, the 'P' signal has not been reset so the fans should re-start. B is correct, IAW the schematic diagram. C is incorrect, the SI signal has no input to the fans start logic. D is incorrect, if a 'P' signal is present the fans will start when the DG breakers close. There are no stepping relay contacts in the fan start logic.
Direct/New/ Modified	New
K/A #:	027K2.01
K/A Values:	3.1/3.4
Cognitive Level:	Memory (I)
References:	Lesson 1120, OBJ L1120I04RO Schematic diagram 1-NHY-310931 sh. BB5 & sh. BC3.

Difference between B) & D)?

Too detailed knowledge needed to exclude D)?

Question Number:	SRO 69
Question:	Plant conditions: • Containment pressure is 15 psig. • The crew is performing step #8 of FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, which checks containment hydrogen concentration. Why are the hydrogen recombiners <u>NOT</u> placed in service if hydrogen concentration is 4% or more? A. This concentration is well below the lower flammability limit of hydrogen and recombiner operation is not required. B. This concentration is at the explosive limit for hydrogen and a containment air purge is the preferred method for reducing hydrogen concentration. C. Recombiner operation is not required as there is no relationship between high containment pressure and a containment challenge due to hydrogen burn. D. Recombiner operation could cause a hydrogen burn resulting in a pressure spike, which may challenge Containment integrity.
Answer:	D
Justification:	A is incorrect as H2 concentration of 4% or more is potentially flammable. B is incorrect as 4% H2 concentration is not close to the lower explosive limit for hydrogen concentration. C is incorrect as the opposite is true. D is correct in accordance with the background information for Step #8 of FR-C.1
Direct/New/ Modified	New
K/A #:	028K5.02
K/A Values:	3.4/3.9
Cognitive Level:	Memory (I)
References:	Background document for FR-C.1, Step #8 Objective L1206I10RO

Question Number:	SRO 70
Question:	Plant conditions: • MODE 1, 100% power • D7251, CONTAINMENT PURGE PRESSURE HIGH alarms on VA2 • Containment pressure on COP-PI-1787 indicates 15.4 psia. • A review of plant log information indicates that this appears to be a normal containment pressure rise due to hot summer weather conditions. What action should be taken by the crew? IAW proced. A. Place COP in service to reduce containment pressure and clear the alarm. B. Place CAP in service to reduce containment pressure and clear the alarm. C. Place the non-running Containment Structure Cooling Fan in service to reduce containment pressure and clear the alarm. D. No action is required. Containment pressure is within the Technical Specification limits of 14.6 to 16.2 psia.
Answer:	A
Justification:	A is correct IAW VAS procedure for D7251. COP is placed in service to reduce containment pressure to 15.2 psia. B is incorrect, CAP is not used for minor adjustments in containment pressure. C is incorrect, only 5 of 6 CAH fans are normally running. The 6th fan unit would not be run for minor containment pressure adjustment. D is incorrect, while it is true that containment pressure is within Tech Spec limits, action is required by procedure to reduce containment pressure to ensure margin to the tech spec limit.
Direct/New/ Modified	New
K/A #:	029A1.03
K/A Values:	3.0/3.3
Cognitive Level:	Memory (I)
References:	VPRO D7251 Objective L1120I15RO

Needed to know from memory?

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Question Number:	SRO 71
Question:	The plant is in MODE 6 with CORE ALTERATIONS in progress. RM-6527A-1, TRN A channel 1, Containment Building Purge Line monitor has been declared INOPERABLE. How does this impact CORE ALTERATIONS? A. CORE ALTERATIONS may continue indefinitely because the Containment Building Purge Line monitors are only required in MODES 1 through 4. B. CORE ALTERATIONS may continue indefinitely provided the CAP and COP Containment penetrations are closed. C. CORE ALTERATIONS may continue for up to 7 days provided the CAP and COP Containment penetrations are closed. D. CORE ALTERATIONS must be suspended until RM-6527A-1 has been returned to service.
Answer:	A
Justification:	A is correct because TS table 3.3-6 identifies applicable MODES for RM-6527 as 1-4. TS 4.9.9 states that purge and exhaust valves must close on initiation of CVI from the manipulation crane monitor. B is incorrect because CAP and COP are not required to be isolated. C is incorrect because of the stated AOT and the CAP/COP valves are not required to be closed. D is incorrect because CORE ALTERATIONS do not have to be suspended.
Direct/New/ Modified	New
K/A #:	034K6.02
K/A Values:	3.3
Cognitive Level:	Memory (I)
References:	TS table 3.3-6 and TS 4.9.9 Lesson L1142I, Objective L1142I10RO

~~Importance of monitoring?~~

~~Spelled out~~
Familiar to operators

Question Number:	SRO 72
Question:	The plant is at 20% power. SG 'A' MSIV inadvertently CLOSES What will be the INITIAL effect on the listed parameters for the 'A' SG? SG Level / SG Pressure / Loop Tcold A. INCREASE INCREASE DECREASE B. DECREASE INCREASE INCREASE C. DECREASE DECREASE DECREASE D. INCREASE DECREASE INCREASE
Answer:	B.
Justification:	With MSIV closure, the affected SG NR level will shrink. Because there is heat input from the RCP but no heat removal, pressure will go up. Because there is no heat removal until ASDV's or safety valves open, RCS Tcold increases approaching T hot in the affected loop.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	035K6.01
K/A Values:	3.2/3.6
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1405I, Objective L1405I02RO

Question Number:	SRO 73
Question:	Which of the following correctly identifies a procedure transition from the EOP network to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE? A. E-3, STEAM GENERATOR TUBE RUPTURE, when isolation of the ruptured SG from the intact SGs to be used for cooldown is unsuccessful. B. FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, when CETCs are greater than 1100°F and actions to cool the core are unsuccessful. C. FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE, when containment pressure has exceeded 52 psig and neither train of CBS is operating. D. ECA-0.0, LOSS OF ALL AC POWER, when intact steam generators cannot be depressurized to reduce RCS leakage.
Answer:	B
Justification:	B is correct IAW the symptoms or entry conditions for SACRG-1. None of the other answers identifies a valid transition from the EOP network to the SAMGs.
Direct/New/ Modified	New
K/A #:	2.4.16
K/A Values:	3.0/4.0
Cognitive Level:	Memory (I)
References:	FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, STEP #20. SACRG-1, SEVER ACCIDENT CONTROL ROOM GUIDLINE INITIAL RESPONSE, symptoms or entry conditions. Objective L1206I10RO

Question Number:	SRO 74
Question:	Which of the following describes the operation of the Emergency bus second level undervoltage protection scheme? A. When 1 of 2 relays sense bus voltage less than 95% of nominal for 1.2 seconds (RAT available), it initiates a sequence of load stripping and subsequent bus reenergization by the DG. B. When 1 of 2 relays sense bus voltage drop below 25% of nominal, they initiate auto closure of the RAT supply breaker. C. When both relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), they initiate a sequence of load stripping and subsequent bus reenergization by the DG. D. When both relays sense bus voltage less than 95% of nominal coincident with an SI existing for greater than 10 seconds, they initiate a sequence of load stripping and subsequent bus reenergization by the DG.
Answer:	D
Justification:	A, B, & C are incorrect as none of these answers provide the correct coincidence AND correct setpoint for 2nd level U/V protection. D is correct IAW 1-NHY-310102 sh A53h.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	062K4.01
K/A Values:	3.2
Cognitive Level:	Memory (I)
References:	Detailed Systems Text, 4.16KV Distribution System, pages 21 & 22. Schematic Diagram, 1-NHY-310102 sh A53h. Objective L1093I13RO

Question Number:	SRO 75
Question:	DG-1A has been placed in LOCAL control. How will DG-1A respond to a Loss of Off-Site Power? A. DG-1A will automatically start, the output breaker will automatically close, and load sequencing will automatically occur. B. DG-1A will automatically start, the output breaker must be manually closed, and load sequencing will automatically occur upon breaker closure. C. DG-1A must be manually started, the output breaker must be manually closed, and load sequencing must be performed manually. D. DG-1A will automatically start, the output breaker must be manually closed, and load sequencing must be performed manually.
Answer:	B.
Justification:	B is correct per OS1200.02, Attachment C, Caution 1, and the DG system text. On loss of power, the DG will automatically start whether in Local or Remote control. The output breaker must be closed, but when it is closed, sequencing will occur automatically. A is incorrect because the output breaker will not close automatically. C is incorrect because the DG does not have to be manually started. D is incorrect because load sequencing does not have to be performed manually.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	064A4.01
K/A Values:	4.0/4.3
Cognitive Level:	Application (III)
References:	OS1200.02, Attachment C DG system text Lesson Plan L1100I, Objective L1100I16RO

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Question Number:	SRO 76
Question:	From the list of PROCESS Radiation Monitors below, SELECT the monitors that have an associated control actuation function. 1. R-6509, Waste Liquid Test Tank discharge monitor 2. R-6519, Steam Generator Blowdown Flash Tank discharge monitor 3. R-6514, Waste Liquid Test Tank Inlet monitor 4. R-6505, Condenser Air Evacuation discharge monitor 5. R-6516, PCCW Loop A Activity monitor A. 1 and 2 ONLY B. 1, 2, and 3 ONLY C. 2 and 3 ONLY D. 1, 2, 4, and 5 ONLY
Answer:	B.
Justification:	Refer to RM system description, table 4.3 for monitors with control actuations. A is incorrect because it is incomplete with regard to the provided list. C is also incomplete. D includes 2 monitors (#4 and 5) that do not have control actuations
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	073K1.01
K/A Values:	3.9
Cognitive Level:	Memory (I)
References:	RM system text, table 4.3 Objective L1141I06RO

Question Number:	SRO 77
Question:	Plant conditions: • A reactor startup was in progress with reactor power at approximately 2%. • A PT-507 failure caused the main steam dump valves to open. • The BOP operator closed the steam dumps using the steam dump interlock control switches. • RCS Tave is 545°F and slowly increasing What ACTION must be taken in accordance with Technical Specifications? A. RCS temperature must be restored to 551°F within 15 minutes. B. RCS temperature must be restored to 551°F within the next hour. C. RCS temperature must be restored to 557°F within the next 15 minutes. D. RCS temperature must be restored to 557°F within the next hour.
Answer:	A
Justification:	Only answer A describes the correct combination of RCS temperature and ACTION time limit in accordance with T.S. 3.1.1.4, MINIMUM TEMPERATURE FOR CRITICALITY, 557°F is no-load (0% power) RCS Tave.
Direct/New/ Modified	New
K/A #:	2.1.33
K/A Values:	3.0/4.0
Cognitive Level:	Memory (I)
References:	T.S. 3.1.1.4, MINIMUM TEMPERATURE FOR CRITICALITY Various LOIT objectives that require candidate to apply 1 hour or less Tech Spec ACTIONS from memory.

Question Number:	SRO 78
Question:	Which of the following describes the operation of the Service Air isolation valves, SA-V92 and SA-V93, during an Instrument Air leak? A. AUTOMATICALLY CLOSE at 90 psig decreasing, resets to allow MANUAL OPENING above 93 psig INCREASING. B. AUTOMATICALLY CLOSE at 90 psig decreasing, AUTOMATICALLY REOPEN above 93 psig INCREASING. C. AUTOMATICALLY CLOSE at 80 psig decreasing, resets to allow MANUAL OPENING above 83 psig INCREASING. D. AUTOMATICALLY CLOSE at 80 psig decreasing, AUTOMATICALLY REOPEN above 83 psig INCREASING.
Answer:	A
Justification:	Lesson Plan L1104I states that SA-V92 and SA-V93 close <90 psig and reset >93 psig. The reset allows manual opening via control switch. B is incorrect because SA-V92 and V93 do not auto open when pressure is regained. C and D are incorrect because the setpoints listed are incorrect. Additionally, D is also incorrect because the valves do not automatically open.
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	078K4.02
K/A Values:	3.2 / 3.5
Cognitive Level:	Memory (I)
References:	Lesson Plan L1104I, Objective L1104I05RO and L1104I14RO

Question Number:	SRO 79
Question:	A fire alarm actuates at FP-CP-451 in the East pipe chase. The Unit Supervisor enters OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION, and dispatches the fire brigade to investigate. The brigade leader confirms an actual fire and commences fire-fighting actions. OS1200.00 requires that the control mode selector switches for the 'B' & 'C' ASDVs be placed in the CLOSE position. Spell out What is the purpose of this action? A. Prevents spurious "Hot Short" operation of the ASDVs. B. Assures personnel safety by preventing the operation of the ASDVs while the fire brigade is fighting the fire. C. Forces overpressure protection to the steamline safety valves, which are "Fire Rated" components. D. The off-normal position of the switches reminds control room personnel of fire fighting action in the pipe chase.
Answer:	A
Justification:	A is correct in accordance with appendix R of the UFSAR. B, C, & D are incorrect. While this action prevents the ASDVs from actuating none of the reasons listed in these distracters is the reason that the ASDVs are disabled.
Direct/New/ Modified	New
K/A #:	2.4.27
K/A Values:	4.1
Cognitive Level:	Comprehension (II)
References:	OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION. Seabrook UFSAR, appendix R requirements. Objective L1191I04RO

Better distracter

Determination of causation — comprehension

Challenge Memory?

Question Number:	SRO 80
Question:	The plant is in MODE 5. Train B RHR is in service in COOLDOWN mode. The following conditions exist: • Tave is 182°F and STABLE • RH heat exchanger outlet valve, RH-HCV-607 is 10% OPEN • RH heat exchanger bypass flow control valve, RH-FCV-619, is maintaining total RHR flow at 3500 gpm A loss of Instrument Air occurs. Which of the following describes the effect on the RH system and on RCS temperature? RH-HCV-607 / RH-FCV-619 / RCS Temperature A. FAILS AS IS / FAILS AS IS / INCREASES B. FAILS AS IS / FAILS CLOSED / INCREASES C. FAILS OPEN / FAILS AS IS / DECREASES D. FAILS OPEN / FAILS CLOSED / DECREASES
Answer:	D.
Justification:	A and B are incorrect because HCV 607 is a fail-open valve. C is incorrect because FCV-619 is a fail closed valve. D is correct because each valve will fail in the safe position, directing full flow through the RH heat exchanger. The increased flow through the heat exchanger will result in an RCS cooldown
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	005K4.10
K/A Values:	3.1/3.1
Cognitive Level:	Analysis (III)
References:	RH system text, section 3.1.3 Lesson Plan L1115I, Objective L1115I07RO

Question Number:	SRO 81
Question:	The plant is at 100 % power. The PCCW system is aligned for automatic operation. With no immediate operator action, which of the following describes an effect of the Train 'A' PCCW heat exchanger temperature controller failing to the "FULL FLOW" MODE of operation? A. PCCW flow to the Letdown Heat Exchanger will increase. Capitalize B. RCP 'A' Motor Bearing temperature will INCREASE. C. RCP 'A' Thermal Barrier cooling isolation valve, V-428, will CLOSE on high flow. D. The temperature of letdown flow leaving the Letdown Heat Exchanger will DECREASE.
Answer:	D
Justification:	A is incorrect because the Letdown Heat Exchanger will receive roughly the same flow at a lower temperature Dangerous B is incorrect because RCP 'A' gets cooling from PCCW 'A', which is at a lower temp. with full flow. C is incorrect because the TBCW isolation will close on high flow due to a leak. The cooler PCCW will not cause a high flow condition. D is correct because the Letdown TCV is normally fully closed due to "design leakage". When PCCW temperature decreases the Letdown Heat Exchanger will remove more heat from letdown, causing temperature to decrease.
Direct/New/ Modified	Direct from 1998 NRC exam **NOTE: D was modified from original after validation based upon actual system operating characteristics.**
K/A #:	008K3.01
K/A Values:	3.4/3.5
Cognitive Level:	Analysis (III)
References:	OS1212.01, PCCW system malfunction, Operator action summary Lesson Plan L1118I, Objective L1118I08RO

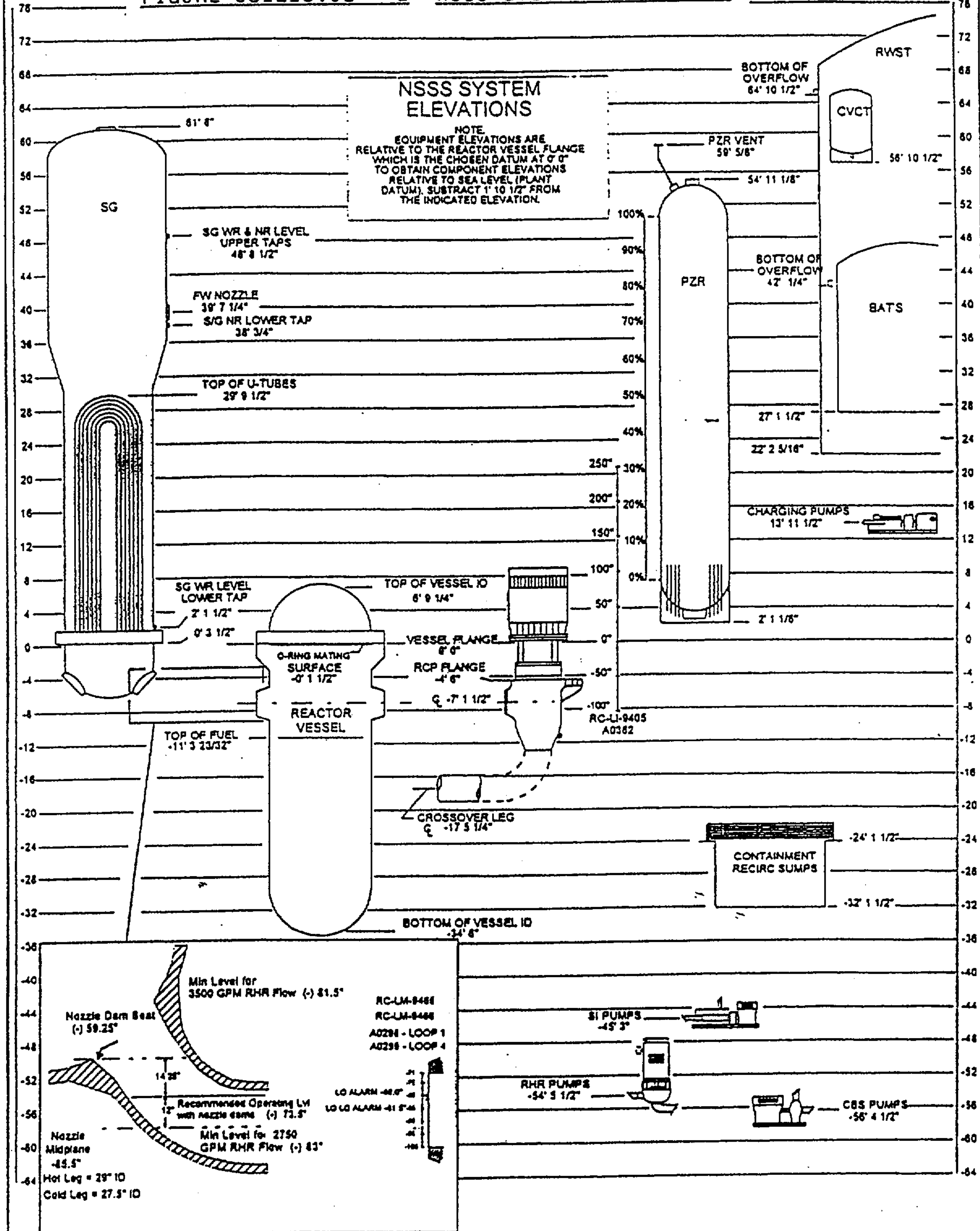
Question Number:	SRO 82
Question:	Plant conditions: • The operating crew has initiated an RCS cooldown at step # 14 of E-3, STEAM GENERATOR TUBE RUPTURE. • Steam dumps are in the steam pressure mode of control. • Steam dump pressure controller PK-507 is in MANUAL at 40% output. • As RCS Tave approaches 550°F the BOP operator holds both STM DUMP P12 BLOCK/P12 BYPASS INTERLOCK control switches in the BYPASS position until the STM DUMP INTERLOCK BYPASSED lamp illuminates on UL-26. Which of the following describes the expected status of the steam dump system? A. The Group 1 "Cooldown" valves are fully open and the group 2 valves are partially open. B. The Group 1 "Cooldown" valves are fully open. C. The Group 1 "Cooldown" valves are partially open. D. All steam dump valves are fully closed.
Answer:	B
Justification:	A is incorrect, bypassing the P-12 Lo-Lo Tavg steam dump block only allows the Group 1 'Cooldown' valves to open in response to the pressure controller, all other groups are isolated. B is correct, C & D are incorrect, a 25% output signal on PK-507 will fully open the Group 1 valves. Since the controller is at 40% output the Group 1 valves will be full open.
Direct/New/ Modified	New
K/A #:	041A1.02
K/A Values:	3.1/3.2
Cognitive Level:	Analysis (III)
References:	Lesson plan 1129, OBJ L1129I12RO

Question Number:	SRO 83
Question:	Plant conditions: • The crew was transferring the 'B' train of service water from the cooling tower back to the ocean. • The 'B' Service Water pump and the 'B' Cooling Tower pump were running. • A Loss of Offsite Power (LOP) occurred and both emergency busses have been reenergized from the emergency diesel generators. • The discharge valve for the 'B' Cooling Tower pump failed to close. Which of the following 'B' train pumps, if any, will be started by the sequencer? A. None. B. The 'B' Cooling Tower pump. C. The 'B' Service Water pump. D. The 'D' Service Water pump.
Answer:	A
Justification:	SW-P-110B discharge valve (SW-V25) must be full closed for the pump to be started by the EPS at step 8 (B incorrect). The EPS will attempt to start the ocean SW pump that was previously running (D incorrect). The 'B' ocean SW pump cannot be started by the EPS as this pump also has a start interlock that requires SW-V25 to be closed (C incorrect). As a result none of the 'B' Train pumps can be started by the EPS (A correct).
Direct/New/ Modified	Direct from bank
K/A #:	076K4.02
K/A Values:	3.2
Cognitive Level:	Memory (I)
References:	1-NHY-301107, Sh.AU6, AR3, & AR4- Lesson Plan 1119, Objective L1119I06RO

Question Number:	SRO 84
Question:	Which of the following is a general responsibility of the Unit Supervisor? IAW A. Responsible to function as the STA, if qualified, if the Shift Manager is absent or <u>not</u> qualified. B. Responsible for notifying management and regulatory agencies as required by station reporting and notification requirements. C. Responsible for unit operations being conducted in accordance with approved station orders, procedures, and Technical Specifications. D. Responsible for initiating call out of personnel to fill vacant shift operating positions.
Answer:	C
Justification:	C is correct IAW OPMM chapter 1, section 3.4. A & D are responsibilities of the work control supervisor. B is the responsibility of the shift manager.
Direct/New/ Modified	New
K/A #:	2.1.2
K/A Values:	3.0/4.0
Cognitive Level:	Memory (I)
References:	OPMM, Lesson L1505I, OBJ L1505I06RO

Question Number:	SRO 85
Question:	Plant conditions: • The plant has been shutdown for 10 days • RCS temperature is 120°F • RCS water level is at the (-)73.5 inch elevation with nozzle dams installed • The crew has just experienced a loss of the running RHR pump and has entered OS1213.02, LOSS OF RHR WHILE OPERATING AT REDUCED INVENTORY OR MID-LOOP CONDITIONS. • A representative from Reliability and Safety Engineering is not available in the outage "War Room" Using the attached reference material, determine the minimum time to initiation of boiling based on plant conditions. A. 21 minutes B. 17 minutes C. 13 minutes D. 10 minutes ie boiling will be occurring at t=21 minutes
Answer:	B
Justification:	Using figure OS1213.02-4, TIME TO BOILING VS. TIME AFTER SHUTDOWN FOR RCS AT MID-LOOP CONDITIONS, and entering chart for 240 hours at initial RCS temperature of 120°F the closest answer is 17 minutes. Candidate must determine which figure in the procedure to use and must know that mid-loop conditions are defined in OS1000.12 to be water level less then –71 inches.
Direct/New/ Modified	New
K/A #:	2.1.25
K/A Values:	3.1
Cognitive Level:	Analysis (III)
References:	Figure OS1213.02-04, TIME TO BOILING VS. TIME AFTER SHUTDOWN FOR RCS AT MIDLOOP CONDITIONS Objectives L1705I05RO & L1705I08RO

FIGURE OS1213.02 - 1 NSSS SYSTEM ELEVATIONS

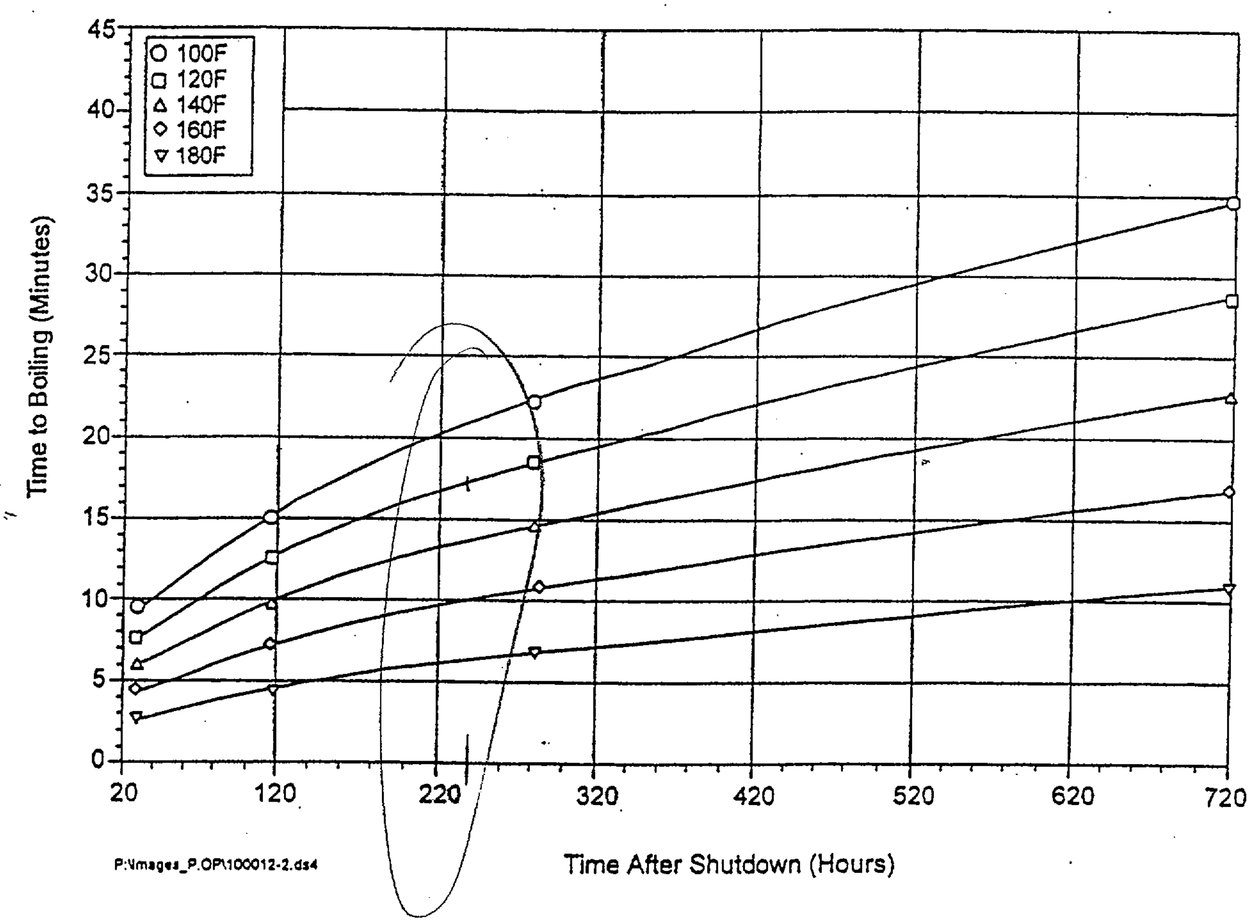


35a

FIGURE OS1213.02-4
TIME TO BOILING vs. TIME AFTER SHUTDOWN FOR RCS WATER LEVEL AT MIDLOOP

CAUTION
- The best estimate of time to boil is provided by Reliability and Safety Engineering (War Room). The graph below provides conservative time to boil values and should only be used if no other information is available.
- For planning purposes, the RCS time to boil should be based on the assumption of atmospheric pressure even when the RCS is evacuated.

NOTE The temperatures(e.g. 100°F,120°F,140°F,etc.) for different curves refer to initial RCS temperature at the time that loss of RHR cooling occurs. All curves assume atmospheric pressure at the onset of boiling.



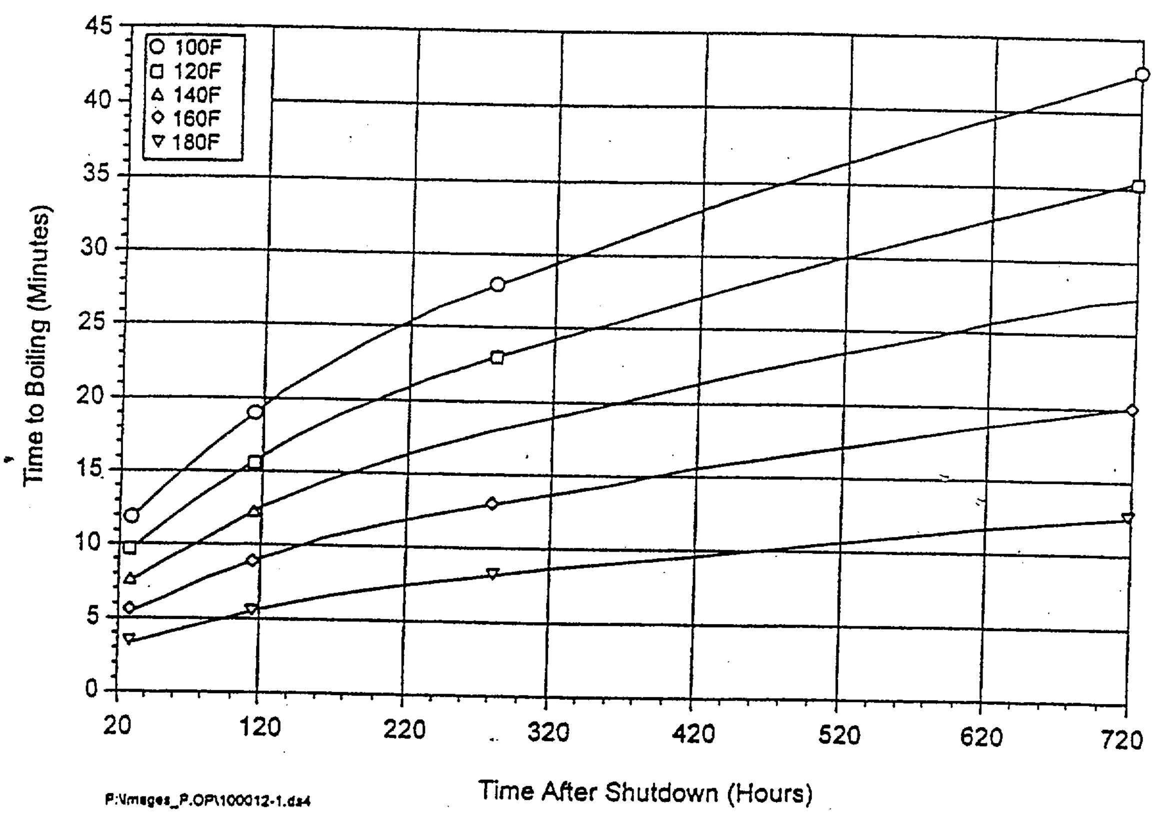
85b

FIGURE OS1213.02-5
TIME TO BOILING vs. TIME AFTER SHUTDOWN FOR RCS WATER LEVEL AT REDUCED INVENTORY (MINUS 36 INCHES)

CAUTION

- The best estimate of time to boil is provided by Reliability and Safety Engineering (War Room). The graph below provides conservative time to boil values and should only be used if no other information is available.
- For planning purposes, the RCS time to boil should be based on the assumption of atmospheric pressure even when the RCS is evacuated.

NOTE The temperatures(e.g. 100° F,120° F,140° F,etc.) for different curves refer to initial RCS temperature at the time that loss of RHR cooling occurs. All curves assume atmospheric pressure at the onset of boiling.



85a

Question Number:	SRO 86
Question:	In accordance with ODI 45, SYSTEM LINEUP PERFORMANCE, which of the following describes how to conduct an Independent Verification if the component to be verified is in a High Radiation Area? from A. The Independent Verification may be performed form a distance if it can be verified as being the proper component. B. The Independent Verification may be waived providing there is remote indication of the component to be verified. C. The Independent Verification may be performed concurrently with the initial repositioning. D. The Independent Verification may be marked as N/A by the US on the system lineup.
Answer:	C
Justification:	ODI 45 states the following: "5.2.2.1 Performing system lineups and independent verifications in high radiation or contaminated areas can be done concurrently to minimize exposure." C is correct. D is incorrect, OS1090.05 states the SM/US may waive the IV to minimize exposure, however, the waiver is entered in the Unit Journal and documented on the Verification Form. A & B are incorrect, there is no mention in the ODI or in OS1090.05 about using distance or remote indication to perform the IV.
Direct/New/ Modified	Direct from bank
K/A #:	2.1.29
K/A Values:	3.4/3.3
Cognitive Level:	Memory (I)
References:	ODI 45, OS1090.05 section 3.11 Objective L1505I17RO

Question Number:	SRO 87
Question:	The crew is preparing to perform a rapid plant shutdown from 100% power using Figure 6: Rapid Power Decreases Guidelines, of OS1000.06, POWER DECREASE. The Unit Supervisor directs the PSO to turn on the 'A' and 'B' pressurizer Backup Heaters to force pressurizer spray. What is the reason for this direction? A. Ensures proper mixing occurs such that RCS loop boron samples accurately reflect actual boron concentration. B. Prevents boron stratification in the pressurizer spray nozzles. C. Prevents an RCS dilution event from occurring during a pressurizer outsurge. D. Ensures adequate flow through the pressurizer to prevent boron precipitation on the pressurizer heaters.
Answer:	C
Justification:	A is incorrect because PZR spray operation ensures proper mixing in the PZR and resultant representative PZR boron samples not RCS loop boron samples. B is incorrect because this phenomenon is fictitious but plausible, i.e. sounds similar to RCS loop temperature stratification. C is correct because pressurizer heater operation causes continuous PZR spray which ensures a minimum boron concentration differential between the PZR and the RCS loops precluding an inadvertent boron dilution event as a result of a PZR outsurge of water into the RCS. D is incorrect, boric acid will not precipitate on the PZR heaters as long as the heaters are covered with water.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	2.1.32
K/A Values:	3.4/3.8
Cognitive Level:	Memory (I)
References:	OS1000.07, APPROACH TO CRITICALITY INPO SOER 94-02, Boron Dilution Events Objective L1167I02RO

Question Number:	SRO 88
Question:	The plant reduced power from 100% to 15%, dumping steam to the main condenser via the steam dumps. While lined up to perform a hotwell discharge, seawater was vacuum dragged into the hotwell via a leaking check valve. Chemistry has confirmed that a seawater intrusion has occurred. The condensate pump cation conductivity is now indicating 1.2 μmhos. Which of the following ~~is the required course~~ of action per OS1234.02, CONDENSER TUBE ~~OR TUBE~~ OR TUBE SHEET LEAK? Typo A. Reduce reactor power to ~5% power. Establish feed flow to the SGs using the SUFP taking a suction on the CST. Continue hotwell cleanup using OS1234.02. B. Reduce reactor power to ~5% power. Establish feed flow to the SGs using the SUFP taking a suction on the hotwells. Simultaneously cleanup the hotwells using OS1234.02. C. Perform a reactor shutdown within 1 hour. Establish feed flow to the SGs using the SUFP taking a suction on the CST. Continue hotwell cleanup using OS1234.02. D. Trip the reactor perform E-0, REACTOR TRIP OR SAFETY INJECTION. Establish feed flow to the SGs using the SUFP taking a suction on the CST. Continue hotwell cleanup using OS1234.02. Do plant conditions make a difference wrt the course of action?
Answer:	D
Justification:	D is the correct response per OS1234.02. Continued power operation is not allowed making A, B, & C wrong.
Direct/New/ Modified	New
K/A #:	2.1.34
K/A Values:	2.9
Cognitive Level:	Application (III)
References:	OS1234.02, CONDENSER TUBE OR TUBE SHEET LEAK. Objective L1188I03RO

Question Number:	SRO 89
Question:	Which of the following is the required re-evaluation frequency for Temporary Modifications (TMODs) installed per MA4.3, TEMPORARY MODIFICATIONS? A. 60 days B. 90 days C. 120 days D. 180 days
Answer:	B
Justification:	Section 4.6 specifically sets the re-evaluation frequency at 90 days.
Direct/New/ Modified	New
K/A #:	2.2.11
K/A Values:	3.4
Cognitive Level:	Memory (I)
References:	MA 4.3, section 4.6 Objective L1510I24SR

Any extention?

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 90
Question:	Plant conditions: • It is Monday morning at 0805 • The plant is in MODE 1 • OX1446.01, AC POWER SOURCE WEEKLY OPERABILITY, was last performed 7 days ago (at 1500). • The TRN 'A' Emergency Diesel Generator was declared INOPERABLE five minutes ago (0800). Which of the following identifies the <u>latest</u> time that the crew can complete OX1446.01 and still meet applicable Technical Specification ACTION requirements? A. 0900 today B. 1500 today C. 1600 today D. 0900 Wednesday
Answer:	A
Justification:	A is correct because TS 3.8.1.1 ACTION b requires the surveillance to be performed within 1 hour of the DG being declared inoperable. B is incorrect because it is 168 hours from the previous completion, which is not a requirement. C is incorrect because it is 8 hours from the time the DG was declared inoperable. D is incorrect because it reflects 168 hours (7 days) plus 25% extension, which is not applicable.
Direct/New/ Modified	New (Modified from 1998 Salem SRO exam)
K/A #:	2.2.12
K/A Values:	3.4
Cognitive Level:	Application (III)
References:	TS 3.8.11, OX1446.01 Objective L1100I24RO

Question Number:	SRO 91
Question:	While work is being performed on Feed water heater drain components, the CONTACT PERSON for one work package requests a temporary lift. There are two additional work packages assigned to the clearance. The TAGGING AUTHORITY is unable to locate the CONTACT PERSON on one work package. How should the request be processed? IAW A. Obtain concurrence from another SRO licensed individual and designate an alternate CONTACT PERSON for notification of the temporary lift, then approve the request after notification is made. B. Do NOT approve the request until the designated CONTACT PERSON is located. C. Identify another Level 1 individual who will assume the CONTACT PERSON responsibilities, then approve the request for temporary lift. D. Temporarily assign the person requesting the temporary lift the CONTACT PERSON responsibility for all work packages under the clearance while the temporary lift is in effect. When responsibility has been assumed, approve the request.
Answer:	C
Justification:	A is incorrect because another SRO licensed individual is not required. B is incorrect because the procedure provides for assigning an alternate CONTACT PERSON. C is correct as outlined in sections 4.6 and 4.9 of MA 4.2. D is incorrect because the new CONTACT PERSON is NOT required to assume the responsibility for all work packages under that clearance.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	2.2.13
K/A Values:	3.8
Cognitive Level:	Memory (I)
References:	MA 4.2, section 4.5 Lesson Plan L1501I, Objective L1501I17RO

Question Number:	SRO 92
Question:	The Containment On-line Purge System (COP) is being placed in service for Containment entry. ~~to perform maintenance on the TRN 'A' Containment Instrument Air compressor.~~ Which of the following is used for configuration control when the setpoints for the COP Radiation monitors are changed? A. OS1023.69, CONTAINMENT ON-LINE PURGE SYSTEM OPERATION. B. MA 4.3, TEMPORARY MODIFICATIONS. C. MA 4.4, CONFIGURATION CONTROL DURING MAINTENANCE AND TROUBLESHOOTING. D. MA 4.6, RDMS DATABASE ITEM CONTROL.
Answer:	D
Justification:	A is incorrect because OS1023.69 directs the operators to use MA 4.6. B is incorrect because RDMS setpoint per MA 4.6 are specifically excluded from the scope of MA 4.3. C is incorrect because MA 4.5 is not used when configuration changes are made using approved procedures. D is correct per OS1023.69.
Direct/New/ Modified	New
K/A #:	2.2.14
K/A Values:	3.0
Cognitive Level:	Memory (I)
References:	OS1023.69, MA 4.3, MA 4.5 Objectives L1141I10RO

Connections?

If a) refers to d) then it could be acceptable

Memory of procedures

Follow up Next week

Caps

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 93
Question:	The crew is responding to the following alarm: D4057-HTR 26B EX CHK VLV 5 FULL CLOSED One of the steps in the VPRO is to verify that the extraction check valve is closed. Using the attached drawing, select the statement that describes the indication the operator should have when checking the valve closed. A. Red light on the back of the MCB lit and white light on UL 15 not lit. B. Red light on the back of the MCB not lit and white light on UL 15 lit. C. Red light on the back of the MCB and white light on UL 15 both lit. D. Red light on the back of the MCB and white light on UL 15 both not lit.
Answer:	C
Justification:	1-NHY-503537 identifies C as the correct answer. A, B & D are all different combinations of wrong answers.
Direct/New/ Modified	Direct from facility bank
K/A #:	2.2.15
K/A Values:	2.9
Cognitive Level:	Application (III)
References:	1-NHY-503537

Too Basic ?

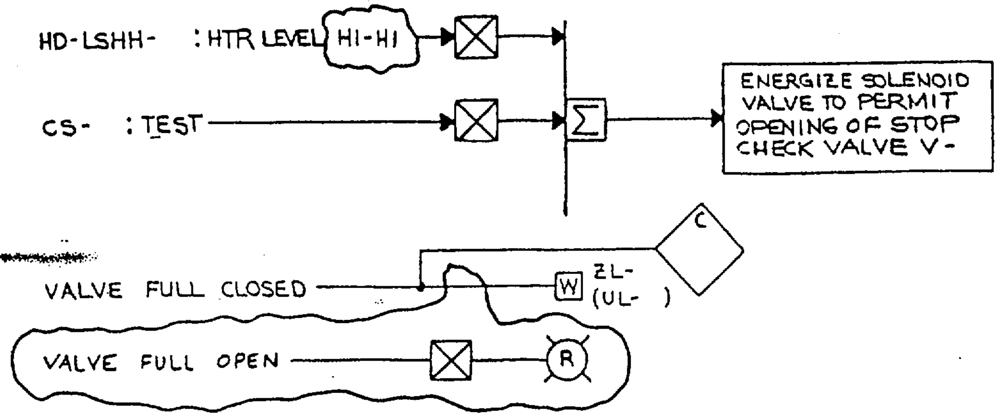
DWG. TRANSFERRED TO CUSTODY OF NHY AT REV. 3 LTR. SPL[R] A121 DTD. 10-21-86

ISSUED-FOR-CONSTRUCTION

EX-NON RETURN VALVES V-8 & V-11 LOGIC DIAGRAM

New Hampshire Yankee Seabrook Station

1-NHY-503537 REV:

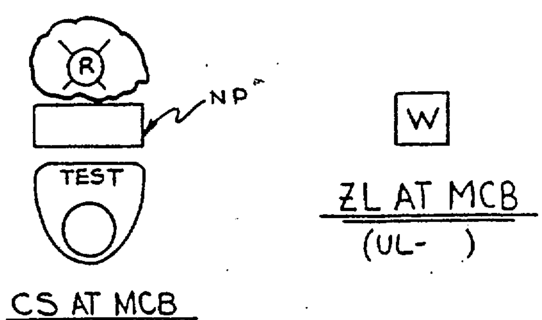


RECORDS MANAGEMENT DEPT. CONTROL NUMBER 5011

EQPT/SERVICE	VALVE	HD-LSHH-	CS-	ZL-	UL-15	CPTR ID NO RTU
FW HTR CO-E-25A	V11	4476	3325	3325	98	D4058 / 8
25B	V8	4477	3326	3326	102	D4059 / 9
FW-E-26A	V2	4495	3327	3327	106	D4056 / 8
FW-E-26B	V5	4496	3328	3328	110	D4057 / 9

REV	DATE	DRWN	CHKD	CE	LOE	DESCRIPTION
6	7/21/94	MRB	MRJ	7/21/94	CM	INCORP MMOD 94-524 CA1
5	12/14/88	HP	AMR	APL	JFG	INCORP. DCR 87-215, CA-0
4	5/4/87	HK	BCE	TAN	N/A	9763-M-503537 SUPERCEDES UE&C DWG:

RECORDS MANAGEMENT DEPT CONTROL NUMBER 402



NOTE
1- FOR DESCRIPTION OF LOGIC SYMBOLS SEE M-503100

REFERENCE DOCUMENTS:

M-506446
M-506447
M-506448
M-506449
9D-1G

1-NHY-503537 REV: 6

93a

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 94
Question:	A non-licensed operator on shift has reached 3000 mrem TEDE exposure for the current year. Which of the following correctly states whose permission is required for the operator to receive additional exposure? A. The Operations Manager and a Health Physics Supervisor. B. The Operations Manager and the Health Physics Department Manager C. A Health Physics Supervisor and the Health Physics Department Manager. D. The Health Physics Department Manager and the Station Director.
Answer:	C
Justification:	C is correct per figure 5.3 of RP-5.1. A & B are incorrect because the Operations Manager does not approve upgrades. D is incorrect because the Station Director approval is required only to exceed 4000 mrem.
Direct/New/ Modified	New
K/A #:	2.3.2
K/A Values:	2.9
Cognitive Level:	Memory (I)
References:	RP-5.1, Objective L1525I13RO

SRO role ?

Question Number:	SRO 95
Question:	To prevent an unscheduled plant shutdown, it has been determined that a contractor will be dispatched into Containment under a Planned Special Exposure (PSE). In addition to the Health Physics Department Manager and the Station Director, who must approve the PSE? A. The individual being exposed B. The Shift Manager (SM) C. The Health Physics Supervisor D. The individual's employer
Answer:	D
Justification:	Direct from RP 5.2 section 3.4 A is incorrect because the procedure does not require the worker to volunteer. B is incorrect because the SM may initiate a PSE cover form but does not approve a PSE. C is incorrect because the HP supervisor initiates the RWP but does not approve the PSE.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	2.3.4
K/A Values:	3.1
Cognitive Level:	Memory (I)
References:	RP5.2 section, 3.0 & 4.0 Objective L1525I14SR

Is Station Director required to have SRO license

Question Number:	SRO 96
Question:	Which of the following describes the flowpath for performing an air purge of Containment to reduce hydrogen concentration while in the emergency operating procedures? A. From the service air system into Containment via the normal H_2 analyzer sample lines. Out of Containment via CGC-V14 and CGC-V28 to the Containment enclosure emergency exhaust filters and then out the plant vent. B. From the service air system via normally locked and closed valves into Containment. Out of Containment via CGC-V14 to the inlet of the Train 'A' Containment enclosure emergency exhaust filter and then out to PAH-F-16 C. From the service air system via normally locked closed valves into Containment. Out of Containment via CGC-V14 and CGC-V28 to the inlet of the Containment enclosure emergency exhaust filters and then out the plant vent. D. From the service air system via normally locked open valves through a Containment isolation check valve into Containment. Out of Containment via CGC-V14 or CGC-V28 to the Containment enclosure emergency exhaust filter to PAH-F-16.
Answer:	C
Justification:	A is incorrect because the hydrogen analyzer sample lines are not used for this purpose. B is incorrect because PAH-F-16 is not the alignment flowpath. Purged Containment air goes to the inlet of <u>both</u> Containment Emergency Exhaust Filters and then out the plant vent. C is correct per the normal operating procedure. D is incorrect because the CGC system service air supply isolation valves to Containment are locked <u>closed</u> valves and PAH-F-16 is not in the flowpath.
Direct/New/ Modified	Direct from 1996 NRC exam
K/A #:	2.3.9
K/A Values:	2.5/3.4
Cognitive Level:	Memory (I)
References:	OS1023.72, AIR PURGE OF CONTAINMENT. Objective L1120I09RO

B is a subset of C ?

Question Number:	SRO 97
Question:	A Steam Generator Tube Rupture has occurred. RCS pressure control has been lost and the crew transitions to ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL. Which of the following reflects the order of steps performed in attempting to restore RCS pressure control? A. First, try to establish normal PZR spray. Then, try to restore a PZR PORV. Finally, try to establish auxiliary spray. B. First, try to establish auxiliary spray. Then, try to restore a PZR PORV. Finally, try to establish normal PZR spray. C. First, try to restore a PZR PORV. Then, try to establish normal PZR spray. Finally, try to establish auxiliary spray. D. First, try to establish normal PZR spray. Then, try to establish auxiliary spray. Finally, try to restore a PZR PORV.
Answer:	A
Justification:	A is the order presented in the procedure. B, C, & D are incorrect variations.
Direct/New/ Modified	New
K/A #:	2.4.6
K/A Values:	3.8
Cognitive Level:	Memory (I)
References:	ECA-3.3 Objective L1212I05RO

Always high press position?

Expected to know from memory?

Question Number:	SRO 98
Question:	The plant is at 75% power when the following alarms are received: • D4433 L-TOP TRAIN B ARMED • D4434 L-TOP A 100# FROM ACTUATION Which of the following is the appropriate Abnormal Operating Procedure to enter in response to this condition? A. OS1201.06, PT-455-458 PZR PRESSURE INSTRUMENT FAILURE, due to Pressurizer pressure channel P456 failing high. B. OS1201.08, TAVG DELTA T INSTRUMENT FAILURE, due to RCS loop2 Tc failing low. C. OS1201.09, RCS WIDE RANGE PRESSURE OR TEMPERATURE INSTRUMENT FAILURE, due to loop 2 wide range Th failing low. D. OS1247.01, LOSS OF A VITAL INSTRUMENT PANEL PP-1A, PP-1B, PP-1C, PP-1D, due to loss of power to PP-1B
Answer:	C
Justification:	A is incorrect because failure low of pressurizer channel P456 will not generate the stated conditions. B is incorrect because RCS loop narrow range temperature channels do not feed PORV control logic. C is correct because auctioneered low wide range Thot is an input to PORV control logic. D is incorrect because loss of PP-1B would result in the opposite TRN alarms.
Direct/New/ Modified	New
K/A #:	2.4.11
K/A Values:	3.6
Cognitive Level:	Analysis (III)
References:	OS1201.09 Objective L1182I07RO

[illegible] choice

Question Number:	SRO 99
Question:	While performing ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, the operator is directed to feed the ruptured steam generator when indicated NR level decreases to ≤ 14%. If ruptured steam generator pressure decreases in an uncontrolled manner the operator is directed to stop feeding the ruptured steam generator. Which of the following describes the basis for stopping feed flow? A. Uncontrolled depressurization is indicative of a faulted steam generator and feed flow is stopped to avoid a PTS concern in the RCS. B. Uncontrolled depressurization can be caused by steam condensation due to overfeeding the ruptured steam generator and feed flow is stopped to prevent reinitiation of break flow. C. Uncontrolled depressurization can be caused by steam condensation around uncovered U-tubes and feed flow is stopped to prevent continued shrinking of NR level. D. Uncontrolled depressurization can be caused by loss of RCS pressure control and feed flow is stopped to prevent unborated water from entering the RCS.
Answer:	B
Justification:	A is incorrect because potential for PTS is minimized with the plant cooldown and depressurization steps of E-3. B is correct per the ES-3.1 background document. C is incorrect because feeding should not cause shrink with the conditions presented in ES-3.1. D is incorrect because the intent of ES-3.1 is to backfill the RCS from the ruptured S/G to facilitate ruptured S/G cooldown. Introduction of unborated water into the RCS is already accounted for in the procedure.
Direct/New/ Modified	Direct from facility bank
K/A #:	2.4.18
K/A Values:	3.6
Cognitive Level:	Memory (I)
References:	ES-3.1, Background document Objective L1205I08RO

Question Number:	SRO 100
Question:	A SITE AREA EMERGENCY was declared 50 minutes ago. Notification has been made to the States and the NRC. Conditions have stabilized and the event can be terminated. Who is responsible for termination of the classification? A. Short Term Emergency Director B. Emergency Operations Manager C. Licensing Coordinator D. Response Manager
Answer:	D.
Justification:	A is incorrect because notifications have been made and the event was not cleared before notifications were made. (ER 1.2, Precaution 3.5) B is incorrect because the Emergency Operations Manager is responsible for Operations personnel. C is incorrect because the Licensing Coordinator coordinates regulatory interface. D is correct per ER1.2C. Either the Response Manager or the Site Emergency Director are responsible for termination
Direct/New/ Modified	Direct from 1998 NRC exam
K/A #:	2.4.40
K/A Values:	4.0
Cognitive Level:	Memory (I)
References:	ER 1.2C, #14 Lesson Plan L1509I, Objective L1509I01SR

Modified questions with original

Question Number:	SRO 03
Question:	During performance of OX1410.02, QUARTERLY ROD OPERABILITY CHECK AND MONTHLY NEW FULL OUT POSITION SURVEILLANCE, Rod F2 in Control Bank 'B', group 1 stops moving when it is 14 steps from it's new "Full Out" bank position. I&C reports that the Lift Coil has failed and the rod is declared INOPERABLE. Technical Specification 3.1.3.1, ACTION b.3.d limits reactor power to 75% Rated Thermal Power. Which of the following is the reason for this power limit? A. Acceptable power distribution is assured and continued operation is allowed if the rod is declared untrippable. B. Allows the plant to be operated without performing a re-evaluation of the safety analysis affected by a misaligned rod. C. Relieves the operators of having to calculate shutdown Margin every 12 hours. D. Provides assurance of fuel rod integrity during continued operations.
Answer:	D
Justification:	A is incorrect because a 6 hour shutdown is required per action a if the rod is declared untrippable. B is incorrect because the T.S. <u>does</u> require a reevaluation of the safety analysis affected by a misaligned rod. C is incorrect because the T.S. requires the Shutdown Margin be calculated every 12 hours until the rod is realigned. D is correct from the basis for T.S 3.1.3.1.
Direct/New/ Modified	Modified from 1996 SRO exam
K/A #:	000005K3.04
K/A Values:	4.1
Cognitive Level:	Analysis (III)
References:	Technical Specifications 3.1.3.1 and its associated basis.

QUESTION: 041

Given the following:

- ONE control rod is misaligned from its group by more than twelve (12) steps and determined to be INOPERABLE.
- The Technical Specification ACTION statement limits reactor power to 75% Rated Thermal Power.

Which of the following is the reason for this power limit?

a. Reduces the Rod Insertion Limit below the misaligned rod position.

b. Allows the plant to be operated without performing a re-evaluation of the safety analysis affected by a misaligned rod.

c. Relieves the operators of having to calculate Shutdown Margin every 12 hours.

d. Provides assurance of fuel rod integrity during continued operations.

ANSWER:

d.

KA RATING	RO	SRO
APE005 K3.04		4.1

REFERENCES:

Technical Specifications 3.1.3.1 and its associated basis.

OBJECTIVES:

L1113I22RO

JUSTIFICATION:

Answer a is incorrect because the reduction of power has no effect on the value of the RIL for a given power.

Answer b is incorrect because the T.S. <u>does</u> require a reevaluation of the safety analysis affected by a misaligned rod.

Answer c is incorrect because the T.S requires the Shutdown Margin be calculated every 12 hours until the rod is realigned.

Question Number:	SRO 05
Question:	The following plant conditions exist: • A rupture in the piping downstream of SI-V138/139 has occurred • The check valves on the piping connecting to the RCS have failed causing a LOCA into the Containment penetration area of the PAB • The Reactor has tripped and Safety Injection has actuated on Low PZR Pressure • All ECCS systems are operating as designed. • The crew transitions to ECA-1.2, LOCA OUTSIDE CONTAINMENT, at step # 25 of E-0 based upon abnormal PAB radiation level. Assuming plant conditions do not significantly change and the leak is unisolable, what procedure in the EOP network will ultimately be used to deal with this accident? A. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION. B. ECA-1.2, LOCA OUTSIDE CONTAINMENT C. E-1, LOSS OF REACTOR OR SECONDARY COOLANT. D. ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.
Answer:	D
Justification:	A is incorrect as there are no procedural transition criteria to ES-1.2 in this condition. B is incorrect as ECA-1.2 contains actions that assume LOCA outside containment is in the RHR system. This is not the case and ECA-1.2 will direct transition to ECA-1.1. C is incorrect as question stem indicates that LOCA outside containment is unisolable and therefore a transition to E-1 would not be made. D is correct. The crew should transition to ECA-1.1 from ECA 1.2, Step 4 RNO due to inability to isolate leak. ECA-1.1 contains the long term actions required to cooldown and depressurize the RCS when emergency coolant recirculation is unavailable.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	E04EK1.3
K/A Values:	3.9
Cognitive Level:	Analysis (III)
References:	E-0, Step 25 ECA-1.2, entry conditions and step 4 ECA-1.1, entry condition #3 Objective L1209I03SR

QUESTION: 006

The following plant conditions exist:

- A rupture in the piping downstream of the Charging Cold Leg injection Valves (SI-V138/139) has occurred
- The check valves on the piping connecting to the RCS have failed causing a LOCA into the Containment penetration area of the PAB
- The Reactor has tripped and Safety Injection has actuated on Low PZR Pressure
- The Crew has entered E-0 and has completed the Immediate Actions
- RCS pressure is 1780 psig and decreasing
- Area and airborne high radiation alarms have actuated in the PAB
- All ECCS systems are operating per design
- Containment sump level is 0 feet on LI-2384 and LI-2385

Assuming plant conditions do not significantly change and the leak is unisolable, what is the expected flow path through the EOP Network for this accident?

a. E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant" to ES-1.2, "Post LOCA Cooldown & Depressurization".

b. E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant" to ECA-1.2, "LOCA Outside Containment" to ECA-1.1, "Loss of Emergency Coolant Recirculation".

c. E-0, "Reactor Trip or Safety Injection" to ECA-1.2, "LOCA Outside Containment" to E-1, "Loss of Reactor or Secondary Coolant".

d. E-0, "Reactor Trip or Safety Injection" to ECA-1.2, "LOCA Outside Containment" to ECA-1.1, "Loss of Emergency Coolant Recirculation".

ANSWER:

d.

KA RATING	RO	SRO
WEST E04 EK1.3		3.9

REFERENCES:

E-0, Step 32
ECA-1.2, Entry Conditions and Step 4
ECA-1.1, Entry Condition #3

OBJECTIVES:

L1209I03SR

JUSTIFICATION:

Answers a. and b. are incorrect because conditions for transitioning to E-1 from E-0, Step 24 will not be met because the LOCA is in the PAB.

Answer c. is incorrect because the leak will be unisolated using the actions of ECA-1.2. Step 5 of ECA-1.2 will have the Crew transition to ECA-1.1.
Answer d. is correct. The crew should transition to ECA-1.2 from E-0, Step 30 due to high Aux. Bldg. radiation levels. (Note: ECCS termination criteria will not be met at Step 25 of E-0 due to decreasing RCS pressure). Based on the break location the actions of ECA-1.2 will be unsuccessful in isolating the leak. The crew should transition from ECA-1.2, Step 4 RNO to ECA-1.1.

Question Number:	SRO 06
Question:	A small break LOCA has occurred. Automatic SI is actuated but the reactor does not trip. In accordance with FR-S.1, the crew shuts the reactor down using manual rod insertion and emergency boration. The emergency boration is continuing. The crew transitions to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, from E-0, Step #20. At Step #8 of E-1 the Primary Operator is directed to reset SI to enable stopping of the RHR pumps. SI will <u>NOT</u> reset. What is a possible cause of the SI reset failure? A. The initiating condition causing the SI actuation has not cleared. B. The Reactor Trip Breakers are closed. C. The timer in the Safety Injection Block/Reset logic has not timed out. D. The automatic reactor trip signal has cleared.
Answer:	B.
Justification:	A is incorrect because SI may be reset as long as the 60 second timer has timed out and P-4 is energized, whether the initiating condition is still active or not. C is incorrect because the procedure flowpath to get to ES-1.1 would be much longer than 60 seconds. D is incorrect because at least one automatic trip signal will remain active in these conditions and in any case the Rx trip breakers never opened. B is correct as P-4 inputs to the SI reset circuitry.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	E02EK2.1
K/A Values:	3.4/3.9
Cognitive Level:	Analysis (III)
References:	IS system text, figure 3.4 RP system text, section 4.4.1.1, page 28 Lesson Plan L1138I, Objective L1138I21RO

Question Number:	SRO 55
Question:	An ATWS has occurred. The crew has shut down the reactor using manual rod insertion and boration. One PORV has stuck open on the initial pressure transient, resulting in Safety Injection actuation. The stuck open PORV has been ISOLATED. The crew has transitioned to ES-1.1, SI TERMINATION. When the Primary Board Operator attempts to reset SI, it will not reset. What is a possible cause of the SI reset failure? A. The initiating condition causing the SI actuation has not cleared. B. The Reactor Trip Breakers are closed. C. The timer in the Safety Injection Block/Reset logic has not timed out. D. P-11 is not energized.
Answer:	B. The reactor trip breakers are closed
Justification:	A is incorrect because SI may be reset as long as the 60 second timer has timed out and P-4 is energized, whether the initiating condition is still active or not. C is incorrect because the procedure flowpath to get to ES-1.1 would be much longer than 60 seconds. D is incorrect because P-11 does not have to be energized to allow SI reset. P-11 provides the ability to block the Low Pressurizer Pressure SI at pressures below 1950 psig. Additionally, P-11 will cause acumulator isolation valves to open if they are in Auto above the P-11 setpoint
Direct/New/ Modified	New
K/A #:	E02EK2.1 Instruments, signals, interlocks, failure modes, automatic and manual features
K/A Values:	3.4/3.9
Cognitive Level:	Analysis
References:	IS system text, figure 3.4 RP system text, section 4.4.1.1, page 28 Lesson Plan L1138I, Objective L1138I21RO

Question Number:	SRO 07
Question:	The plant is at 30 % power during a Plant Startup with all control systems in AUTOMATIC. RCP 'A' trips on Phase Differential Overcurrent. Assuming no operator action, which of the following describes the response of the plant to the RCP trip? A. Steam flow decreases in all SGs. All SG levels initially decrease, then increase as the secondary plant stabilizes and SGWLC responds. Control Rods withdraw to maintain Tave on program. B. Steam flow decreases in all SGs. SG 'A' level initially increases due to overfeeding. SG 'B', 'C', 'D' levels initially decrease due to increased steam demand. SG levels return to normal as SGWLC responds. Tave remains unaffected because Reactor power remains unaffected. C. Steam pressure decreases in all SGs. SG 'A' level decreases due to the loss of heat input. SG 'B', 'C', 'D' levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Tave and Tref stabilize at a lower value. D. Steam pressure decreases in all SGs. SG 'A' level decreases due to the loss of heat input. SG 'B', 'C', 'D' levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Control rods withdraw to return Tave to it's previous value.
Answer:	C
Justification:	A is incorrect because Steam flow increases on the 'B', 'C', & 'D' SGs to make up for the loss of steam flow from 'A' SG. Also 'A' SG NR level will initially decrease from shrink affect. B is incorrect because the SG levels will respond in an opposite manner to that described because of shrink (A SG) and swell (B, C, D SGs), and steam flow decreases from the 'A' SG. C is correct. Tref will decrease as Pimp decreases with Tsat in the SGs. Rod control will function to restore Tave to the new lower programmed value. D is incorrect. Rod control will not restore Tave to it's previous value because Tref has decreased, changing the Tave program (it will be a lower temperature).
Direct/New/ Modified	Modified from 1998 NRC exam.
K/A #:	000015/000017AK1.04 2.1.7
K/A Values:	2.9/3.1 3.7/4.4
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1405I Abnormal Transient Analysis Objectives L1405I01RO and L1405I02RO

Question Number:	SRO 35
Question:	The plant is at 12 % power during a Plant Startup. The following VAS alarms are acknowledged: • D5779 RCP A MOTOR FRAME VIBRATION HIGH • D9775 RCP A SHAFT VIBRATION HIGH The crew stops the load increase to investigate. While the crew is investigating the cause of the vibration alarm, RCP A trips on Phase Differential Overcurrent. With no operator action, describe the response of the plant to the RCP trip. A. Steam flow decreases in all SGs. All SG levels initially decrease, then increase as the secondary plant stabilizes and SGWLC responds. Control Rods withdraw to maintain Tave on program. B. Steam pressure decreases in all SGs. SG A level initially increases due to overfeeding. SG B, C, D levels initially decrease due to increased steam demand. SG levels return to normal as SGWLC responds. Tave remains unaffected because Reactor power remains unaffected. C. Steam pressure decreases in all SGs. SG A level decreases due to the loss of heat input. SG B,C,D levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Tave stabilizes at a lower value. D. Steam pressure decreases in all SGs. SG A level decreases due to the loss of heat input. SG B,C,D levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Control rods withdraw to maintain Tave on program.
Answer:	C. Steam pressure decreases in all SGs. A SG level decreases due to the loss of heat input. B,C,D SG levels increase due to increased steam demand. SG levels return to normal as SGWLC responds. Tave stabilizes at a lower value.
Justification:	See attached description. A and D are incorrect because at 12% power, control rods have not yet been placed in auto. B is incorrect because the SG levels will respond in an opposite manner to that described because of shrink (A SG) and swell (B,C,D SGs). Also, Tave will decrease because Tc decreases in the

	operating loops as Delta T in those loops increases.
Direct/New/ Modified	New
K/A #:	000015/000017AK1.04 Steady state relationship between RCS and SG with a failed RCP
K/A Values:	2.9/3.1
Cognitive Level:	Comprehension
References:	Lesson Plan L1405I Abnormal Transient Analysis Objectives L1405I01RO and L1405I02RO

Question Number:	SRO 08
Question:	The plant is operating at 100 % power when a Loss of Off-Site power causes a reactor trip. Two minutes following the trip, the following conditions exist: • All 4 S/G pressures trending slowly upward toward ASDV lift setpoint. • Core Exit Thermocouple temperatures are slowly increasing. • RCS Cold leg temperatures are slowly increasing. • RCS Hot leg temperatures are slowly increasing. Based on the above indications, what is the condition of the RCS? A. Natural Circulation has developed. Heat removal is being maintained by the condenser steam dumps. B. Natural Circulation has not developed. Heat removal may be established by opening the condenser steam dumps. C. Natural Circulation has developed. Heat removal is being maintained by atmospheric steam dumps. D. Natural Circulation has not developed. Heat removal may be established by opening the atmospheric steam dump valves.
Answer:	D
Justification:	A and B are incorrect because on a loss of off site power, condenser steam dumps are unavailable due to loss of circulating water pumps. C is incorrect based on expected indications for natural circ flow in Appendix A of ES-0.1. D is correct because question stem conditions indicate that natural circ has not developed based on criteria in appendix A of ES-0.1. Dumping steam with ASDVs will help establish natural circ cooling.
Direct/New/ Modified	Modified from 1998 NRC Exam
K/A #:	E09EK2.2
K/A Values:	3.6/3.9
Cognitive Level:	Synthesis (III)
References:	ES-0.1, appendix A Lesson Plan L1200I, Objective L1200I08RO

Question Number:	SRO 40
Question:	The plant is operating at 100 % power when a Loss of Off-Site power causes a reactor trip. Ten minutes following the trip, the following conditions exist: • SG A Pressure 1135 psig and stable • SG B Pressure 1125 psig and stable • SG C Pressure 1130 psig and stable • SG D Pressure 1125 psig and stable RCS Pressure is 2250 psi and stable Thot is approximately 580°F in all 4 loops and trending down slowly Core Exit TC's indicate approximately 585°F Tcold is approximately 561°F in all 4 loops and stable Based on the above indications, what is the condition of the RCS? A. Natural Circulation exists. Heat removal is being maintained by the condenser steam dumps. B. Natural Circulation does not exist. Heat removal may be established by opening the condenser steam dumps. C. Natural Circulation exists. Heat removal is being maintained by atmospheric steam dumps. D. Natural Circulation does not exist. Heat removal may be established by opening the atmospheric steam dump valves.
Answer:	C. Natural Circulation exists, maintained by atmospheric steam dumps
Justification:	A and B are incorrect because on a loss of off site power, condenser steam dumps are unavailable due to loss of circulating water pumps. C is correct based on steam table indications, with Tcold approximately equal to the saturation pressure of all 4 SG's and Thot trending down slowly D is incorrect because the stem contains no indication of a lack of or loss of natural circulation
Direct/New/	New

Modified	
K/A #:	E09EK2.2 Facility heat removal operations
K/A Values:	3.6/3.9
Cognitive Level:	Synthesis
References:	ES-0.1, appendix A Lesson Plan L1200I, Objective L1200I08RO

Question Number:	SRO 10
Question:	The following plant conditions exist: • Train 'A' RHR is operating in the shutdown cooling mode. • RCS temperature is 320°F and stable • RCP-1C is operating • MPCS VAS B4787 PCCW HD TK LVL RATE OF CHANGE HIGH is in alarm. • TRN 'A' PCCW HEAD TANK level is decreasing WHICH of the following is the cause of the noted conditions? A. A tube leak in the CVCS regenerative heat exchanger. B. A tube leak in a Seal Water Return heat exchanger. C. A leak in the 'C' RCP thermal barrier heat exchanger. D. A tube leak in the 'A' RHR heat exchanger.
Answer:	B
Justification:	B is correct because the PCCW operating pressure is higher than the seal return pressure. A is incorrect because the PCCW is not supplied to the CVCS regenerative heat exchanger. C & D are incorrect because the RCS pressure required to support RCP operation is greater than the operating pressure of PCCW.
Direct/New/ Modified	Modified from bank
K/A #:	026AA2.01
K/A Values:	3.5
Cognitive Level:	Analysis (III)
References:	P&ID 1-CC-B2020A, CC Objective: L1118I03RO

SAVE

#10

Removed from IEB on 1/12/00 for NRC exam
Restore to IEB after EXAM

~~Question: 19033~~ See attached page

UPDATED: 6/9/98 QUESTION APPROVED: Yes MINUTES TO ANSWER:6
SYSTEM: CC NRC CATEGORY: 19
PHASE: ~~REQUAL~~
ATTACHMENTS: False

OBJECTIVES:
L3105C 05 ~~RO~~
~~L3105C 07~~ RO

L1118I03RO

Question
The following plant conditions exist:

- RCS temperature is 320°F on Train A RHR cooling.
- C RCP is operating.
- ~~It is noted that the A PCCW head tank level is decreasing more rapidly than normal~~.
- MPCS VAS B4787 PCCW HD TK B LVL RATE OF CHANGE HIGH alarms.
- TRN PCCW Head Tank level is decreasing

Which one of the following could cause the noted condition

A. A tube leak in the CVCS regenerative heat exchanger.

B. A tube leak in a Seal Water Return heat exchanger.

C. A leak in the C RCP thermal barrier heat exchanger.

D. A tube leak in the A RHR heat exchanger.

Reference P&ID ~~1-CC-B20~~

Answer

B

Question Number:	SRO 12
Question:	Plant Conditions: • A plant event has resulted in implementation of ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS. • Bus E-6 is de-energized leaving the steam driven EFW pump as the only source of EFW flow to the steam generators. • NR level in all four S/Gs is off scale low. • The operators have throttled EFW flow to 25 gpm per S/G in accordance with step #2 of ECA-2.1 causing a RED path on heat sink. • The steam driven EFW pump subsequently trips. What action should the crew take? A. Continue with ECA-2.1. The caution prior to step #1 prohibits transition to FR-H.1. B. Attempt to place the startup feedpump in service while continuing with ECA-2.1. C. Transition to FR-H.1. RESPONSE TO LOSS OF SECONDARY HEAT SINK. D. Continue with ECA-2.1 SG NR levels are adequate.
Answer:	C
Justification:	A is incorrect, the caution prior to ECA-2.1, step #1 states that FR-H.1 should only be implemented if EFW flow capability of 500 gpm is not available. This condition is met and transition to FR-H.1 should be made. B is incorrect, there are no contingency actions in ECA-2.1 to place SUFP in service. C is correct, RED path is valid since no EFW flow is available. D is incorrect, S/G NR levels are NOT adequate as indicated in the stem. Continuation with actions of ECA-2.1 is not appropriate with inadequate EFW flow.
Direct/New/ Modified	Modified from 1996 NRC Exam
K/A #:	E12EA2.2
K/A Values:	3.9
Cognitive Level:	Comprehension (II)
References:	ECA-2.1 FR-H.1 Objective L1207I04RO

QUESTION: 007

A plant event has resulted in implementation of ECA-2.1 'Uncontrolled Depressurization of All Steam Generators".

While attempting to control the RCS cooldown during this procedure, the operator throttles EFW flow which results in a RED path on the Heat Sink Critical Safety Function.

What action should be taken?

a. Raise at least one steam generator narrow range level to greater than 5%, then continue with ECA-2.1.

b. Increase EFW flow to 500 gpm to clear the "RED" condition and continue on in ECA-2.1.

c. Increase EFW flow to 500 gpm to clear the "RED" condition and transition to FR-H.1, "Response to Loss of Secondary Heat Sink".

d. Continue with ECA-2.1.

ANSWER:

d.

KA RATING	**RO**	**SRO**
WEST E-12 EA2.2		3.9

REFERENCES:

ECA-2.1
FR-H.1

OBJECTIVES:

L1207I04RO

JUSTIFICATION:

The CAUTION prior to step #1 in ECA-2.1 tells the operator to implement FR-H.1 only if total feed flow capability of 500 gpm is not available while performing ECA-2.1. In the situation given here, the operator took action to throttle EFW flow but nothing was done to impact the capability of the operator to feed at 500 gpm. Consequently the correct answer is d.

Question Number:	SRO 14
Question:	PLANT CONDITIONS: • Reactor Power is 32 % • Turbine load has been reduced from 900 MWE to 360 MWE • Condenser Vacuum is 23.5 inches Hg and slowly decreasing Based on the above indications, which action is the crew required to take? A. Trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. B. Continue the load decrease to increase condenser vacuum to > 25 inches Hg. C. Trip the turbine and go to ON1231.02, TURBINE TRIP BELOW P-9. D. Remove the turbine generator from service IAW OS1000.06, POWER DECREASE.
Answer:	A
Justification:	A is correct because a manual Rx trip is required when load has been decreased to 360 MWE and vacuum cannot be maintained >25" Hg. B is incorrect because load reduction should not continue below 360 MWE. C is incorrect. A reactor trip is required and power is > P-9 (20%) which will cause a turbine trip. D is incorrect because load was originally > 360 MWE.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	000051AA2.02
K/A Values:	3.9/4.1
Cognitive Level:	Synthesis (III)
References:	Objective L1180I08RO ON1233.01, LOSS OF CONDENSER VACUUM

Question Number:	SRO 48
Question:	PLANT CONDITIONS: • Reactor Power is 37 % • Turbine load is 420 MWE • Condenser Vacuum is 22 inches Hg and stable • Load reduction is in progress • The cause of the vacuum loss has been identified and corrected Based on the above indications, which action is the crew required to take? A. Immediately trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. B. Continue the load decrease to increase condenser vacuum to > 25 inches Hg. C. Immediately trip the turbine, verify all stop valves are closed and the generator breaker opens, and go to ON1231.02, TURBINE TRIP BELOW P-9. D. Continue the load decrease and if vacuum remains greater than 22 inches Hg remove the turbine generator from service IAW OS1000.06, POWER DECREASE.
Answer:	A. Immediately trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION
Justification:	Answer A is correct because a reactor trip is required when a turbine trip is required above P-9 (20 % power) A turbine trip is required because vacuum is below the trip setpoint. Answer B is incorrect because the conditions in the question stem indicate a failure of the turbine trip Answer C is incorrect. Turbine trip is required by the setpoint. However, with power >P-9, a reactor trip is also required Answer D is incorrect because turbine trip is required but did not occur
Direct/New/ Modified	Modified from 1996 LOUT exam question 25
K/A #:	000051AA2.02 Conditions requiring reactor or turbine trip
K/A Values:	3.9/4.1
Cognitive Level:	Synthesis
References:	Objective L1180I08RO ON1233.01, Loss of Condenser Vacuum

Question Number:	SRO 16
Question:	During his routine rounds in the 'A' train essential switch gear room the secondary NSO notices that the "Reverse Transfer" lamp is lit on static transfer switch EDE-CP-1E. What is the condition of EDE-PP-1E? A. The transfer switch has swapped EDE-PP-1E back to its inverter supply. B. The transfer switch has swapped EDE-PP-1E to its alternate supply. C. The maintenance supply breaker at EDE-PP-1E has tripped open. D. EDE-PP-1E is de-energized.
Answer:	B
Justification:	A is incorrect because the reverse transfer lamp indicates that the transfer switch has swapped the power panel to its alternate power supply. B is correct, lamp indicates that EDE-PP-1E is on alternate (non-inverter) power. C is incorrect because the maintenance supply breaker to the panel is normally open. D is incorrect because the reverse transfer light lit indicates that alternate power supply power is being provided to the power panel and no other indicator/alarms which would implicate a loss of power to the panel are indicated in the question stem.
Direct/New/ Modified	Modified from 1996 NRC Exam
K/A #:	000057AA2.06
K/A Values:	3.7
Cognitive Level:	Memory (I)
References:	120 VAC vital instrumentation distribution System detailed system text. VAS alarm response for D5734 Objective L1098I09RO

QUESTION: 017

The plant is at 100% power when the following VAS alarm is received:

- D5734 Vital UPS 1E INV PWR FUSE BLOWN

Locally at EDE-CP-1E (static transfer switch), the operator observes that the reverse transfer light is lit.

Which of the following events is the likely cause of this alarm/indication?

a. PP-1E is being supplied from it's DC supply.

b. PP-1E has transferred to it's alternate supply.

c. PP-1E maintenance supply breaker has tripped open.

d. PP-1E is de-energized.

ANSWER:

b.

KA RATING	**RO**	**SRO**
APE057 A2.06		3.7

REFERENCE:

120 VAC Vital Instrumentation Distribution System detailed system text
VAS alarm response for D5734

OBJECTIVES:

L1098I09RO

JUSTIFICATION:

Answer a. is incorrect because PP-1E is an AC power panel, it is UPS-I-1E that has a backup DC supply.

Answer c is incorrect because the maintenance supply breaker to the panel is <u>normally</u> open.

Answer d is incorrect because the reverse transfer light lit indicates that alternate power supply power is being provided to the power panel and no other alarms which would be actuated on loss of power to the panel are indicated in the question stem.

Question Number:	SRO 19
Question:	A fire in the Train 'A' Electrical Penetration Area has been confirmed by the Fire Brigade, and the control room crew has entered OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION. Prior to initiating the equipment disabling actions identified in the procedure, the 'A' PZR PORV spuriously opens, causing a Safety Injection actuation. What action should the crew take? A. Continue with the actions of OS1200.00 and close the 'A' PORV block valve. B. Transition to E-0, REACTOR TRIP OR SAFETY INJECTION, to deal with the Safety Injection actuation. C. Transition to OS1200.01, SAFE SHUTDOWN AND COOLDOWN FROM THE MAIN CONTROL ROOM. D. Transition to OS1200.02, SAFE SHUTDOWN AND COOLDOWN FROM THE REMOTE SAFE SHUTDOWN FACILITIES.
Answer:	B
Justification:	A, C, & D are incorrect as the transition to E-0 must be made per the direction on the OAS page of OS1200.00 which states: "If a safety injection occurs during the performance of this procedure, than go to E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1."
Direct/New/ Modified	Modified question from 1996 NRC exam
K/A #:	000067AK3.04
K/A Values:	4.1
Cognitive Level:	Memory (I)
References:	OS1200.00, RESPONSE TO FIRE OR FIRE ALARM ACTUATION, KEY CAUTION

QUESTION: 039

A fire in the Train 'A' Electrical Penetration Area has been confirmed by the Fire Brigade, and the control room crew has entered the appropriate fire response procedure. Subsequently a reactor trip occurs.

The procedure flow path the operating crew will follow is:

a. E-0, "Reactor Trip or Safety Injection", OS1200.00, "Response to Fire or Fire Alarm Actuation", OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone, OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room".

b. OS1200.00, "Response to Fire or Fire Alarm Actuation", OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone, OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room".

c. E-0, "Reactor Trip and/or Safety Injection", ES-0.1, "Reactor Trip Response" and simultaneously OS1200.00, "Response to Fire or Fire Actuation".

d. OS1200.00, "Response to Fire or Fire Alarm Actuation", when directed by OS1200.00 enter E-0, "Reactor Trip or Safety Injection" and return to OS1200.00 after the immediate action steps, OS1200.00A, App. A to Fire Hazards Analysis for Affected Area/Zone.

ANSWER:

b.

KA RATING	RO	SRO
APE067 K3.04		4.1

REFERENCES:

OS1200.00, Response to Fire or Fire Alarm Actuation, Key Caution

OBJECTIVES:

L1191I04RO

JUSTIFICATION:

Answers a. & c. are incorrect because even with a reactor trip the operators are directed to not perform E-0. Seabrook procedures will eventually cover the safe shutdown.

Answer d. is incorrect because no steps in E-0 are performed.

The point of knowledge examined is for the examinee to know that E-0 is not entered even if a Reactor Trip occurs.

Question Number:	SRO 20
Question:	Plant conditions: • The control room has been evacuated due to a fire and the remote safe shutdown facilities have been manned. • Remote safe shutdown system lineups have not yet been initiated. • The Local/Remote switch on Bus E-5 for RH-P-8A is in REMOTE. • SSPS in not defeated. • A valid SI signal has just been received. Which of the following describes the response of RH-P-8A? A. The pump will start and remains running until the "SI" signal is reset, at which time the pump will stop. B. The pump will start and remains running until its associated breaker is tripped locally at the switchgear. C. The pump will not automatically start, but the operator can start/stop the pump using the local control switch at the switchgear. D. The pump will not automatically start, nor can it be started locally due to the RMO lockout.
Answer:	B
Justification:	A is incorrect because the 'SI' signal cannot be reset from RSS panel nor would the pump shutdown if only the 'SI' signal were reset. B is correct, per the schematic diagram. C & D are incorrect because the pump will automatically start on an 'SI' signal with the Local/Remote switch in the REMOTE position.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000068AA1.21
K/A Values:	4.1
Cognitive Level:	Memory (I)
References:	RH system text Logic diagram 1-NHY-503761, RH pump logic diagram. Objective L1115I10RO

QUESTION: 046

Given the following:

- The control room has been evacuated due to a fire.
- All remote safe shutdown system lineups have been completed.
- The local/remote switch on Bus E-5 for RHR pump RH-P-8A is in local
- A valid SI signal has just been received.

Which of the following describes the RHR pump response?

a. The pump will start and remains running until the "S" signal is reset, at which time the pump will stop.

b. The pump will start and remains running until its associated breaker is opened locally.

c. The pump will not automatically start, but the operator can start/stop the pump using the local control switch at the switchgear.

d. The pump will not automatically start, but the operator can start/stop the pump from the Train A remote safe shutdown panel.

ANSWER:

c

KA RATING	RO	SRO
APE068 A1.21		4.1

REFERENCES:

RH system text
Logic diagram 1-NHY-503761, RH pump logic diagram

OBJECTIVES:

L1115I10RO

JUSTIFICATION:

Answer a is incorrect because placing the switch in local at the switchgear removes all automatic features including an automatic start on an "S" signal.

Answer b is incorrect because of the reason given above.

Answer d is incorrect because the pump can only be controlled at the switchgear.

Question Number:	SRO 26
Question:	A Small Break LOCA has occurred. The crew is performing the actions of ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. Containment pressure is 5.2 psig and slowly decreasing. ECCS pumps have been stopped. Normal Charging is aligned. The crew is depressurizing the RCS to minimize subcooling. When the depressurization is stopped, the following conditions exist: • RCS Subcooling is 45°F and DECREASING slowly • Pressurizer Level is 62% and DECREASING slowly Based on these indications, what actions should be taken? A. Establish letdown flow to reduce Pressurizer Level to 5%. B. Manually START ECCS pumps as necessary to increase subcooling. C. REINITIATE Safety Injection and verify all safeguards equipment has actuated. D. Continue with the cooldown to cold shutdown. Control charging flow to maintain Pressurizer level greater than 35%.
Answer:	D.
Justification:	A is incorrect, letdown is not established in ES-1.2. B & C are incorrect, subcooling and pressurizer levels are above the criteria for ECCS reinitiation. Safety injection would not be reinitiated. ECCS pumps would be started individually as needed. D is correct, the subcooling and pressurizer level requirements of step #22 are met. The crew should continue the cooldown and control charging flow to maintain pressurizer level.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	000009EA2.01
K/A Values:	4.2/4.8
Cognitive Level:	Application (III)
References:	ES-1.2, step 22 or Operator action summary (ECCS Reinitiation Criteria) Lesson Plan L1204I, Objective L1204I02RO

Question Number:	SRO 59
Question:	A Small Break LOCA has occurred. The crew is performing the actions of ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. ECCS pumps have been stopped. Normal Charging is aligned. The crew is depressurizing the RCS. When the depressurization is stopped, the following conditions exist: • RCS Subcooling is 37°F and DECREASING • Pressurizer Level is 18% and DECREASING Based on these indications, what actions should be taken? A. ISOLATE Letdown. Check to ensure Pressurizer Level stabilizes above 5%. B. Manually START ECCS pumps as necessary to regain subcooling. C. REINITIATE Safety Injection and verify all safeguards equipment has actuated. D. INCREASE RCS pressure using pressurizer heaters to regain subcooling.
Answer:	B. Manually restart ECCS pumps as necessary to regain subcooling
Justification:	Step 22 or Operator Action Summary page for ES-1.2 states the required action. A is incorrect. Letdown is already out of service in this event. C is incorrect. Reinitiation of SI may result in a higher pressure than necessary for the plant conditions, and RHR will be running again at shutoff head. D is incorrect, because although pressurizer heaters are energized to establish a bubble in the pressurizer, they are not used to repressurize the RCS on loss of subcooling
Direct/New/ Modified	New
K/A #:	000009EA2.01 Actions to be taken based on RCS temperature and pressure
K/A Values:	4.2/4.8
Cognitive Level:	Application
References:	ES-1.2, step 22 or Operator action summary Lesson Plan L1204I, Objective L1204I02RO

Question Number:	SRO 27
Question:	A Plant Trip and Safety Injection has occurred, due to a Steam Generator Fault inside containment. The following conditions exist: • All automatic equipment responds as expected • Containment pressure is 3.2 psig and slowly increasing • RCS pressure is 1750 psig and decreasing • Subcooling margin is 105 degrees F and increasing • Pressurizer level is 22% and decreasing Assuming conditions do not significantly change, in which of the following procedures would you expect to be directed to stop one charging pump? A. In E-2, FAULTED STEAM GENERATOR ISOLATION. B. In E-1, LOSS OF REACTOR OR SECONDARY COOLANT. C. In ES-1.2, POST-LOCA COOLDOWN AND DEPRESSURIZATION. D. In ES-1.1, SI TERMINATION.
Answer:	D
Justification:	A is incorrect, no actions exist in E-2 to shutdown ECCS pumps. B is incorrect, the only ECCS pumps stopped in E-1 are the RHR pumps if RCS pressure is >260# and pressure is stable or increasing. C is incorrect, a transition to ES-1.2 will not be made under conditions given in the stem. D is correct, when the faulted S/G completes blowing down a transition to ES-1.1 will eventually be made from E-1. ES-1.1 directs stoppage of all but one charging pump and re-establishment of normal charging and letdown.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	E02EA2.1 2.4.48
K/A Values:	4.2 3.8
Cognitive Level:	Analysis (III)
References:	E-1, E-2, ES-1.2, ES-1.1 Objectives L1204I03RO

QUESTION: 008

Due to a Small Break LOCA, a Plant Trip and Safety Injection has occurred.

The following conditions exist:

- All automatic equipment responds as expected
- Containment pressure is 3.2 psig and slowly increasing
- Containment High Range Radiation Monitors RM-RI-6576-A & B read ~ 3 R/hr
- RCS pressure is 1750 psig and slowly decreasing
- Subcooling margin is 32 degrees F and slowly decreasing
- Pressurizer level is 22% and slowly decreasing

Assuming conditions do not significantly change, in which of the following procedures would you expect to be directed to stop one charging pump?

a. In ES-0.2, "Natural Circulation Cooldown".

b. In E-1, "Loss of Reactor or Secondary Coolant".

c. In ES-1.2, "Post-LOCA Cooldown and Depressurization".

d. In ES-1.1, "SI Termination".

ANSWER:

c.

KA RATING	RO	SRO
WEST E02 EA2.1		4.2

REFERENCES:

E-1
ES-1.2

OBJECTIVES:

L1204I03RO

JUSTIFICATION:

Answer a. is incorrect, a transition to ES-0.2 would not be made under LOCA conditions.

Answer b. is incorrect, the only ECCS pumps stopped in E-0 are the RHR pumps if RCS pressure is > 260# and pressure is stable or increasing.

Answer c. is correct, The SI termination criteria of E-1 step 6 are not met and a transition to ES-1.1 should not be made. Step #13 of ES-1.2 checks if a charging pump should be stopped.

Answer d. is incorrect as a transition to ES-1.1, SI Termination, should not be made under the given conditions of the question.

Question Number:	SRO 30
Question:	Due to a failure of PZR pressure channel 455, PZR pressure channels 457/456 have been selected for control and backup respectively. Sometime later, channel 457 fails LOW. Which of the following describes the effect, if any, this failure has on PORV operation in the present mode? A. Only PORV 456A is prevented from opening automatically. B. Only PORV 456B is prevented from opening automatically. C. Both PORVs are prevented from opening automatically. D. Both PORVs will open automatically when required.
Answer:	C
Justification:	C is correct in accordance with the note prior to Step #1 of OS1201.06, "PZR Pressure Instrument PT 455/458 Failure." A,B, & D are incorrect in accordance with the process control block diagram.
Direct/New/ Modified	Modified from facility Exam Bank
K/A #:	000027AA1.01
K/A Values:	3.9
Cognitive Level:	Analysis (III)
References:	OS1201.06, PZR PRESSURE INSTRUMENT PT 455/458 FAILURE. (Note prior to step #1) 1-NHY-509026, PZR pressure control process control block diagram. Objective L1182I14RO

Direct from bank - deleted - restore after exam

#30

Question: ~~8011~~ 30

~~UPDATED: 9/20/96~~ QUESTION APPROVED: Yes ~~MINUTES TO ANSWER:4~~
~~SYSTEM: PPLC~~ ~~NRC CATEGORY: 19~~
~~PHASE: REQUAL~~
~~ATTACHMENTS: False~~

~~OBJECTIVES:~~
~~L3106C 04 RO~~
~~L1108C 11 RO~~
~~N1108C 11 RO~~
N1108C 14 RO
L1182I 14 RO

Question FAILURE OF PZR PRESSURE CHANNEL 455

Due to a ~~MCB meter indication problem at 50% power~~, PZR pressure channels 457/456 have been selected for control AND BACKUP RESPECTIVELY. Sometime later, channel 457 fails LOW.

WHICH OF THE FOLLOWING ~~SELECT the statement which~~ describes the effect, if any, this failure has on ~~the automatic operation of the~~ PORV~~s~~ OPERATION in the present mode?

A. Only PORV 456A is prevented from opening automatically.

B. Only PORV 456B is prevented from opening automatically.

C. Both PORVs are prevented from opening automatically.

D. Both PORVs will open automatically when required.

Answer

C.

COGNITIVE LEVEL: ANALYSIS (II)

K/A 000AA1.01 IMPORTANCE SRO - 3.9

SOURCE - MODIFIED FROM FACILITY EXAM BANK

REF: OS1201.06, "PZR PRESSURE INSTRUMENT PT 456/458 FAILURE" (NOTE PRIOR TO STEP #1)
1-NHY-509026 PZR PRESS CONTROL PROCESS CONTROL BLOCK DIAG.

JUSTIFICATION: C IS CORRECT IN ACCORDANCE WITH THE NOTE PRIOR TO STEP #1 OF OS1201.06, "PZR PRESSURE INSTRUMENT PT 456/458 FAILURE". A, B & D ARE INCORRECT IN ACCORDANCE WITH THE PROCESS CONTROL BLOCK DIAGRAM.

Question Number:	SRO 35
Question:	The plant is at 15% power when a total loss of Main Feedwater occurs. The reactor does not trip, and the crew enters FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS. What function, if any, will the ATWS Mitigation System provide under these conditions? A. The ATWS Mitigation System is not armed under these conditions. B. The ATWS Mitigation System will send a start signal to the EFW pumps when 1/4 SG NR levels are less than 5%. C. The ATWS Mitigation System will send a trip signal to the Main Turbine when 2/4 detectors on 1/4 SGs are less than 14%. D. The ATWS Mitigation System will send a start signal to the EFW pumps when 3/4 SG NR levels are less than 5%.
Answer:	A
Justification:	A is correct because the system is armed above 20% power. B is incorrect because the correct actuation logic requires NR level to be below 5% in ¾ SGs. C is incorrect, this is the SSPS actuation logic for the EFW pumps. D is incorrect, while this answer defines the proper actuation logic the ATWS mitigation system will not function because it is below it's arming power level.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000054AA2.03
K/A Values:	4.2
Cognitive Level:	Comprehension (II)
References:	EFW Detailed System Description, Lesson Plan 1127, OBJ L1127I03RO

QUESTION: 019

The plant is at 50% power when a total loss of Main Feedwater occurs. The reactor does not trip, and the crew enters FR-S.1.

What function, if any, will the ATWS Mitigation System provide under these conditions?

a. The ATWS Mitigation System is not armed under these conditions.

b. The ATWS Mitigation System will send a start signal to the EFW pumps when 1/4 SG NR levels are less than 5%.

c. The ATWS Mitigation System will send a start signal to the EFW pumps when 2/4 detectors on 1/4 SGs are less than 14%.

d. The ATWS Mitigation System will send a start signal to the EFW pumps when 3/4 SG NR levels are less than 5%.

ANSWER:

d.

KA RATING	**RO**	**SRO**
APE054 A2.03		4.2

REFERENCES:

EFW Detailed System Description

OBJECTIVES:

L1127I03RO

JUSTIFICATION:

Answer a. is incorrect because the system is armed above 40% power.
Answer b. is incorrect because the correct actuation logic requires NR level to be below 5% in 3/4 SGs.
Answer c. is incorrect, this is the SSPS actuation logic for the EFW pumps.
Answer d. is correct.

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 37
Question:	Which of the following will occur on a loss of Vital DC Bus 11A? A. Both EFW pumps start and the MFRV and bypass valves fail open. B. 'A' train PCCW temperature control and bypass valves fail to their minimum cooling positions (HX bypass). C. The 'A' train P-14 solenoids on the MFRV and MFRV bypass valves are de-energized causing these valves to fail closed. D. The 'A' train P-12 solenoids on the steam dump valves are de-energized causing the steam dumps to fail open.
Answer:	C
Justification:	A is incorrect because the MFRV and MFRV bypass valves will fail closed not open. B is incorrect because PCCW temperature control fails to the full cooling position. C is correct in accordance with the caution prior to Step #1 of OS1248.01. D is incorrect because the P-12 solenoids result in a loss of steam dump capability.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	000058AK3.02
K/A Values:	4.2
Cognitive Level:	Memory (I)
References:	OS1248.01, Loss of Vital 125 VDC Bus, Caution Electrical Drawings Objectives L1186I10RO

QUESTION: 057

Which of the following will occur on a loss of Vital DC Bus 11B?

a. Both EFW pumps start and the MFRV and bypass valves fail open.

b. The steam driven EFW pump starts, however EFW flow can be throttled only with the "B" train throttle valves.

c. The "B" train P-14 solenoids on the MFRV and MFRV bypass valves are deenergized causing these valves to fail closed.

d. The "B" train P-12 solenoids on the steam dump valves are de-energized causing the steam dumps to fail open.

ANSWER:

c

KA RATING	RO	SRO
APE058 K 3.02		4.2

REFERENCES:

OS1248.01, Loss of Vital 125 VDC Bus, Caution
Electrical Drawings

OBJECTIVES:

L1186I10RO

JUSTIFICATION:

Answer a is incorrect because the MFRV and MFRV bypass valves will fail closed not open. The EFW pumps would have started on the subsequent low SG level.

Answer b is incorrect because flow can be throttled with both train of throttle valves since they are motor operated.

Answer d is incorrect because the P-12 solenoids result in a loss of steam dump capability.

Question Number:	SRO 41
Question:	The plant is in MODE 6. You are assigned as the Refueling SRO in containment. • Fuel moves are in progress. • There is a spent fuel assembly in the refueling machine mast. • Refueling Cavity level has been DECREASING at approximately 0.5 inches per minute due to a failed RHR suction relief valve. • Refueling Cavity level is currently 17 feet above the reactor vessel flange and DECREASING. Where do you direct the refueling machine operator to place the spent fuel assembly? A. In any core location B. In the RCCA change fixture C. In the upender in a vertical position D. In the transfer canal with the refueling machine mast fully extended
Answer:	A
Justification:	A is correct because it is the direction provided in the reference. B & C are incorrect because the fuel assembly may be exposed if level continued to decrease. D would only be correct if the fuel assembly could not be moved to a core location.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	000036AA1.04 2.2.29
K/A Values:	3.1/3.7 1.6/3.8
Cognitive Level:	Memory (I)
References:	OS1215.05, Loss of Refueling cavity water Objective L1192I02RO

Question Number:	SRO 94
Question:	The plant is in MODE 6. You are assigned as the Refueling SRO in containment. • Fuel moves are in progress. • There is a spent fuel assembly in the refueling machine mast. • Refueling Cavity level has been DECREASING at approximately 2 inches per minute. • The spent fuel assembly CANNOT be moved to a core location • Refueling Cavity level is DECREASING below the reactor vessel flange Where do you direct the refueling machine operator to place the spent fuel assembly? A. In the transfer canal with the refueling machine mast fully extended B. In the RCCA change fixture C. In the upender in a vertical position D. On the transfer canal floor
Answer:	D. On the transfer canal floor
Justification:	A,B,C all can cause the spent fuel assembly to become uncovered. D is correct by OS1215.05, step 4, RNO 3. The assembly is placed in the transfer canal. If cavity level decreases below the vessel flange, the operator should unlatch it and let it fall to the floor
Direct/New/ Modified	New
K/A #:	2.2.29 Knowledge of SRO fuel handling responsibilities
K/A Values:	3.8
Cognitive Level:	Application/Memory
References:	OS1215.05, Loss of Refueling cavity water No specific facility objective found

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 44
Question:	During a reactor startup the Primary Board Operator is withdrawing shutdown Bank E. Which of the following represents the speed at which the shutdown rods should be moving? A. 32 Steps Per Minute B. 48 Steps Per Minute C. 64 Steps Per Minute D. 72 Steps Per Minute
Answer:	C
Justification:	Manual control of shutdown banks is accomplished at 64 SPM. A is incorrect because no rod group moves at 32 SPM when in MANUAL. A is plausible because the rod speed for shutdown bank E is not displayed on the MCB. B is incorrect because it represents the speed of the control banks in MANUAL. D is the fastest that the control banks will move in AUTO.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	001K4.14
K/A Values:	2.6/2.8
Cognitive Level:	Memory (I)
References:	Lesson Plan L1113I, Section 3.2.3, Objective L1113I04RO

Seabrook SRO Examination
Answer Sheet
Final Draft

Question Number:	SRO 1
Question:	During a reactor startup the Primary Board Operator is withdrawing Control Bank D rods for the approach to criticality. Which of the following represents the speed at which the control rods should be moving? A. 8 Steps Per Minute B. 48 Steps Per Minute C. 64 Steps Per Minute D. 72 Steps Per Minute
Answer:	B. 48 Steps Per Minute
Justification:	Manual control of Control banks is accomplished at 48 SPM. A is incorrect because it represents the slowest speed of the control banks in AUTO. C is the speed that shutdown banks are withdrawn at. D is the fastest that control banks will move in AUTO
Direct/New/ Modified	New
K/A #:	001K4.14 Operating parameters, including proper rod speed
K/A Values:	2.6/2.8
Cognitive Level:	Memory
References:	Lesson Plan L1113I, Section 3.2.3, Objective L1113I04RO

Question Number:	SRO 47
Question:	Plant conditions: • A reactor trip and safety injection has occurred. • All Steam Generator Pressures are DECREASING • Containment temperature, pressure, and humidity are INCREASING • Tave is DECREASING • Containment Pressure is currently NON-adverse For this event, which of the following actuations is designed to prevent the containment from exceeding its design pressure limit? A. Containment Isolation Phase B B. Main Steam Line Isolation C. Containment Isolation Phase A D. Feedwater Isolation
Answer:	B.
Justification:	A is incorrect because Phase B Isolation is designed to isolate remaining containment penetrations not isolated by Phase A. B is correct IAW the reference and the design basis of the MSIS. C is incorrect because Phase A is designed to isolate non-essential process lines that do NOT increase the potential for damage to equipment inside containment. D is incorrect because Feedwater isolation is designed to minimize excessive cooldown and protect the turbine from moisture carryover
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	013A1.02
K/A Values:	3.9/4.2
Cognitive Level:	Analysis (III)
References:	IS system text, sections 3.3.3, 3.3.3.1, 3.3.3.3, and 3.3.4 Lesson Plan L1139I, Objective L1139I08RO

Question Number:	SRO 5
Question:	The plant is at 49% power. An event occurs resulting in the following conditions: • All Steam Generator Pressures DECREASING slowly • Containment temperature, pressure, and humidity INCREASING • Tave DECREASING • Reactor power is INCREASING For this event, which of the following actions is designed to prevent the containment from exceeding its design pressure limit? A. Safety Injection Actuation B. Main Steam Line Isolation C. Containment Isolation Phase A D. Feedwater Isolation
Answer:	B. Main Steam Line Isolation
Justification:	A is incorrect because Safety Injection is required for maintaining core integrity. B is correct IAW the reference and the design basis of the MSIS C is incorrect because CIA is designed to prevent fission product release from the containment D is incorrect because Feedwater isolation is designed to minimize excessive cooldown and protect the turbine from moisture carryover
Direct/New/ Modified	New
K/A #:	013A1.02 Containment pressure, temperature, humidity
K/A Values:	3.9/4.2
Cognitive Level:	Analysis
References:	IS system text, sections 3.3.3, 3.3.3.1, 3.3.3.3, and 3.3.4 Lesson Plan L1139I, Objective L1139I08RO

Question Number:	SRO 50
Question:	Which of the following instruments provide input to Train 'B' of the RCS Subcooling Monitor? A. RCS Wide Range Pressure instrument PT-403 and the average of all core exit thermocouples B. RCS Wide Range Pressure instrument PT-405 and the auctioneered high core exit thermocouple C. RCS Wide Range Pressure instrument PT-403 and the auctioneered high average quadrant temperature D. RCS Wide Range Pressure instrument PT-405 and the auctioneered high average quadrant temperature
Answer:	C
Justification:	A is incorrect because there is no input from an average of "all" CET's. B is incorrect because the subcooling monitor utilizes auctioneered high average quadrant temperature. C is correct according to lesson plan L1140I. D is incorrect because PT-405 inputs to ICCM A.
Direct/New/ Modified	Modified from 1998 NRC exam.
K/A #:	017K4.01
K/A Values:	3.4/3.7
Cognitive Level:	Memory (I)
References:	Lesson Plan L1140I Objective L1140I013RO

Question Number:	SRO 68
Question:	Which of the following instruments provide input to Train A of the RCS Subcooling Monitor? A. RCS Wide Range Pressure instrument PT-403 and the average of all core exit thermocouples B. RCS Wide Range Pressure instrument PT-405 and the auctioneered high core exit thermocouple C. RCS Wide Range Pressure instrument PT-403 and the auctioneered high average quadrant temperature D. RCS Wide Range Pressure instrument PT-405 and the auctioneered high average quadrant temperature
Answer:	D. RCS Wide Range Pressure instrument PT-405 and the auctioneered high average quadrant temperature
Justification:	A is incorrect because there is no input from an average of "all" CET's and PT-403 inputs ICCM B B is incorrect because the subcooling monitor utilizes auctioneered high average quadrant temperature C is incorrect because PT-403 inputs ICCM B D is correct according to Lesson Plan L1140IRO
Direct/New/ Modified	New
K/A #:	017K4.01 Input to subcooling monitor
K/A Values:	3.4/3.7
Cognitive Level:	Fundamental
References:	Lesson Plan L1140I Objective L1140I013RO

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 51
Question:	Plant conditions: • A LOCA is in progress. • Containment pressure currently indicates 16 PSIG and decreasing slowly. • Both trains of CBS are operating. • The Containment pressure recorders indicate that pressure increased to a peak of 21 psig. • All Containment Phase B penetrations are isolated and no safeguards actuation signals have been reset. Which of the following indicates the expected status of Containment cooling systems? A. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the RECIRC MODE. B. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the FILTER MODE. C. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the FILTER MODE. D. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the RECIRC MODE.
Answer:	D.
Justification:	A 'P' signal (Hi-3, 18 psig) will trip Containment Structure Cooling fans and CRDM cooling fans, and start Containment Recirc Fans in the RECIRC Mode. FILTER Mode is used for containment pre-entry. A 'P' signal deenergizes the solenoids in the dampers to cause the system to fail to the recirc MODE
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	022A3.01
K/A Values:	4.1/4.3
Cognitive Level:	Analysis (III)
References:	CHV system text, sections 4.1.3 and 4.1.4 Lesson Plan L1120I, Objective L1120I04RO

Question Number:	SRO 9
Question:	A LOCA is in progress. Reactor Trip and Safety Injection have initiated. All safeguards systems are functioning as designed. Containment Pressure indicates 24 psig and trending down slowly. Which of the following describes the status of Containment Cooling Systems? A. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the FILTER MODE. B. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the FILTER MODE. C. Containment Structure Cooling fans are RUNNING; CRDM Cooling fans are RUNNING; Containment Recirculation fans are operating in the RECIRC MODE. D. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the RECIRC MODE.
Answer:	D. Containment Structure Cooling fans are TRIPPED; CRDM Cooling fans are TRIPPED; Containment Recirculation fans are operating in the RECIRC MODE.
Justification:	A 'P' signal (Hi-3, 18 psig) will trip Containment Structure Cooling fans and CRDM cooling fans, and start Containment Recirc Fans in the recirc MODE. Filter MODE is used for containment pre-entry. A 'P' signal deenergizes the solenoids in the dampers to cause the system to fail to the recirc MODE
Direct/New/ Modified	New
K/A #:	022A3.01 Initiation of safeguards mode
K/A Values:	4.1/4.3
Cognitive Level:	Analysis
References:	CHV system text, sections 4.1.3 and 4.1.4 Lesson Plan L1120I, Objective L1120I04RO

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 54
Question:	OS1235.03, SG LEVEL INSTRUMENT FAILURE, contains the following CAUTION statement prior to step #1: "During operation in manual feedwater control at ≥ 65% power, maintain Steam Generator water level 50% to 70% narrow range." What is the basis for this CAUTION? A. Limits the mass in the SGs with respect to the UFSAR steam break analysis. B. Limits the mass in the SGs in consideration of SG overfill during a Steam Generator Tube Rupture Event. C. Provides adequate mass to ensure iodine partitioning during a Steam Generator Tube Rupture Event. D. Provides adequate mass to maintain heat sink during loss of all AC power.
Answer:	C
Justification:	C is correct in accordance with lesson L1193I. A is incorrect as NR level up to 70% is in excess of that assumed in the steam break analysis. B is incorrect as this action maximizes, not limits, S/G water level. D is incorrect, heat sink is not an initial concern in a loss of all AC power event.
Direct/New/ Modified	Modified from bank
K/A #:	2.4.11
K/A Values:	3.4/3.6
Cognitive Level:	Analysis (III)
References:	OS1235.03, SG LEVEL INSTRUMENT FAILURE, lesson plan Objective L1193I13RO

~~Direct from bank - deleted~~ ~~restore after exam~~

54

Question: 10601

UPDATED: 5/10/91 QUESTION APPROVED: No MINUTES TO ANSWER:0
SYSTEM: ~~FW~~ NRC CATEGORY: 7
PHASE: SIM PHASE I
ATTACHMENTS: False

~~OBJECTIVES:~~
~~L1158I 02 RO~~

Question

~~During operation at ≥ 65% power, operators are cautioned to maintain SG level between 50% and 70% NR when SG level control is in~~ manual. OS1235.03, "SG LEVEL INSTRUMENT FAILURE" CONTAINS THE FOLLOWING CAUTION STATEMENT ~~[illegible]~~ PRIOR TO STEP #1:
"DURING OPERATION IN MANUAL FEEDWATER CONTROL AT ≥ 65% POWER, MAINTAIN STEAM GENERATOR WATER LEVEL 50% TO 70% NARROW RANGE"

≥ APS

Which ~~of the following~~ is the basis for ~~that~~ caution?
THIS CAPS

A. Limits the mass in the SGs WITH RESPECT TO THE UFSAR ~~in consideration of~~ the steam break analysis.

B. Limits the mass in the SGs in consideration of SG overfill during ~~SGTR~~. A STEAM GENERATOR TUBE RUPTURE EVENT.

C. Provides adequate mass to ensure iodine partitioning during ~~SGTR~~. A STEAM GENERATOR TUBE RUPTURE EVENT.

D. Provides adequate mass to maintain heat sink during loss of all AC power.

Answer

C.

K/A 2.4.11 Knowledge of abnormal operating procedures
K/A IMPORTANCE: 3.4 / 3.6

JUSTIFICATION: C IS CORRECT IN ACCORDANCE LESSON L1193I
a IS INCORRECT AS NR LEVEL UP TO 70% IS IN EXCESS OF THAT ASSUMED IN THE STEAM BREAK ANALYSIS NOT LIMITS
b IS INCORRECT AS THIS ACTION MAXIMIZES S/G WATER LEVEL
d IS INCORRECT, HEAT SINK IS NOT AN INITIAL CONCERN IN A LOSS OF ALL AC POWER EVENT

Source: ~~DIRECT FROM~~ MODIFIED FROM BANK

COGNITIVE LEVEL: ~~COMPREHENSION (II)~~ ANALYSIS (III)

REF: OS1235.03, SG LEVEL INSTRUMENT FAILURE, LESSON PLAN

OBJ: L1193I13RO

Question Number:	SRO 56
Question:	Plant conditions: • A loss of offsite power has occurred. • The 'B' Emergency Diesel Generator has failed to start. • The 'A' Emergency Diesel Generator is powering bus E5. Which of the following describes the expected electrical power flowpath to PP-1A, PP-1B, and PP-1F? A. Bus E51→MCC E512→UPS-I-1A→PP-1A Battery B-1B→DC bus 11B→UPS-I-1B→PP-1B Battery B-1B→DC bus 11B→UPS-I-1F→PP-1F B. Battery Charger BC-1A→DC bus 11A→UPS-I-1A→PP-1A Bus E61→MCC E612→UPS-I-1B→PP-1B Bus E61→MCC E612→UPS-I-1F→PP-1F C. Battery B-1A→DC bus 11A→UPS-I-1A→PP-1A Battery Charger BC-1B→DC bus 11B→UPS-I-1B→PP-1B Battery B-1B→DC bus 11B→UPS-I-1F→PP-1F D. Battery Charger BC-1A→DC bus 11A→UPS-I-1A→PP-1A Battery B-1B→DC bus 11B→UPS-I-1B→PP-1B Bus E63→MCC E631→480/120v transformer via Static Transfer switch→PP-1F
Answer:	A
Justification:	A is correct. B is incorrect because UPS-I-1A would not be supplied with DC power under these conditions and PP-1B & 1F would not be supplied with AC power. C is incorrect because PP-1A would not be fed from DC power and no AC power is available to power BC-1B. D is incorrect because power would not be available to alternate AC power supply from Bus E63 and UPS-I-1A would not be supplied with DC power under these conditions
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	063K1.03
K/A Values:	2.9/3.5
Cognitive Level:	Memory (I)
References:	125 VAC Electrical System Detailed System Text Objective L1098I03RO

Question Number:	SRO 57
Question:	Which of the following radiation monitors is both a release path monitor AND has an automatic isolation function associated with it? A. 1GM810, Condenser Air Evacuation. B. 1LM216, SG 'A' Blowdown Line Monitor C. 1LM241, High Range Letdown Activity D. 1LM220, PCCW Loop A
Answer:	B
Justification:	A, C, & D are incorrect because although they are release path monitors, they have no automatic functions associated with them. B is correct as 1LM216 will close SB Flash Tank Outlet Valve SB-CV-6519.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	068A4.04
K/A Values:	3.8/3.7
Cognitive Level:	Memory (I)
References:	OS1252.01, PROCESS OR EFFLUENT HIGH RADIATION, Attachment A Objective L1141I06RO

QUESTION: 076

From the list below, select the radiation monitor that is both a release path monitor and has an automatic action associated with it.

a. 1GM810, Condenser Air Evacuation.

b. 1LM805, Turbine Building Sump.

c. 1GA410, Fuel Storage Building Exhaust.

d. 1NG222, Plant Vent Lo Range Gas.

ANSWER:

b

KA RATING	**RO**	**SRO**
072 A4.01		3.3

REFERENCES:

OS1252.01, Process or Effluent High Radiation, Attachment A

OBJECTIVES:

L1141I06RO

JUSTIFICATION:

Answers a, c, and d are incorrect because although they are release path monitors, they have no automatic functions associated with them

Question Number:	SRO 60
Question:	Plant conditions: • Plant was at 100% power with all control systems in AUTOMATIC when the 'B' MFP tripped, inducing a turbine setback. • The unit supervisor has entered OS1231.03, TURBINE RUNBACK/SETBACK. • D7762 "CNTL BK D INSERTION LIMIT LO-LO" alarms on VA2 What is the significance of this alarm? A. Axial flux difference has entered the "FID Dependent" space. B. The MODE 1 Shutdown Margin Limit may have been exceeded. C. The setback rate has exceeded the capabilities of the Control Rod Drive System. D. A malfunction has occurred in the Rod Control System.
Answer:	B
Justification:	A is incorrect because AFD limits are not necessarily exceeded if the rods are inserted below the RIL. C is incorrect because the alarm setpoint for RIL LO-LO is independent of any rate of turbine load decrease (i.e. if the rods were able to insert faster to keep Tavg on program the RIL would still be exceeded). D is incorrect because this alarm does not indicate a rod control system problem.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	001K5.04
K/A Values:	4.3/4.7
Cognitive Level:	Comprehension (II)
References:	Alarm response procedure for D7762, CNTL BK D INSERTION LIMIT LO-LO OS1202.04, RAPID BORATION, Entry Conditions Objective L1183I09RO

QUESTION: 027

PLANT CONDITIONS:

- The unit has just returned to 100% power following a refueling outage.
- All Shutdown Banks are fully withdrawn.
- Control Banks are withdrawn with Control Bank D at 180 steps.
- Rod Control is operating in AUTO.
- A high stator cooling water temperature causes the turbine to begin running back.

During the runback D7762 "CNTL BK D INSERTION LIMIT LO-LO" is received.

What is the significance of this alarm?

a. The Relaxed Axial Offset Limits have been exceeded.

b. The MODE 1 Shutdown Margin Limit may have been exceeded.

c. The runback rate has exceeded the capabilities of the Control Rod Drive System.

d. The level of turbine runback has been excessive.

ANSWER:

b.

KA RATING	**RO**	**SRO**
001 K5.04		4.7

REFERENCES:

Alarm Response Procedure for D7762, CNTL BK D INSERTION LIMIT LO-LO
FSAR 15.4.8.2 step 1.3

OBJECTIVES:

L1183I09RO

JUSTIFICATION:

Answer a. is incorrect because the Relaxed Axial Offset ("doghouse") Limits are not necessarily exceeded if the rods are inserted below the RIL.

Answer c. is incorrect because the alarm setpoint for RIL Lo-Lo is independent of any rate of turbine load decrease (i.e. If the rods were able to insert faster to keep t-avg on program the RIL would still be exceeded).

Answer d. is incorrect because the level of turbine runback is determined by the stator coolant water return temperature (i.e. the runback will continue until the high stator coolant temperature clears).

Answer b. is correct. The alarm setpoint is rods inserted below the RIL. The Alarm Response Procedure requires that OS1202.04 be performed to borate the RCS until the Lo-Lo Limit condition is cleared.

Seabrook SRO Examination
Work Sheet
Draft

Question Number:	SRO 72
Question:	The plant is at 20% power. SG 'A' MSIV inadvertently CLOSES What will be the INITIAL effect on the listed parameters for the 'A' SG? SG Level / SG Pressure / Loop Tcold A. INCREASE / INCREASE / DECREASE B. DECREASE / INCREASE / INCREASE C. DECREASE / DECREASE / DECREASE D. INCREASE / DECREASE / INCREASE
Answer:	B.
Justification:	With MSIV closure, the affected SG NR level will shrink. Because there is heat input from the RCP but no heat removal, pressure will go up. Because there is no heat removal until ASDV's or safety valves open, RCS Tcold increases approaching T hot in the affected loop.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	035K6.01
K/A Values:	3.2/3.6
Cognitive Level:	Analysis (III)
References:	Lesson Plan L1405I, Objective L1405I02RO

Question Number:	SRO 28
Question:	The plant is at 31% power. SG B MSIV inadvertently CLOSES Assuming the reactor does NOT immediately trip, which of the following describes the INITIAL effect on SG B and its associated RCS Loop? SG Level / SG Pressure / Loop DELTA-T A. INCREASE / INCREASE / DECREASE B. DECREASE / INCREASE / DECREASE C. DECREASE / DECREASE / DECREASE D. INCREASE / DECREASE / INCREASE
Answer:	B. DECREASE INCREASE DECREASE
Justification:	With MSIV closure, the affected SG NR level will shrink. Because there is heat input from the RCP but no heat removal, pressure will go up. Because there is no heat removal until ASDV's or safety valves open, the loop Delta T will go down, with Tc approaching Th.
Direct/New/ Modified	New
K/A #:	035K6.01 MSIV failure
K/A Values:	3.2/3.6
Cognitive Level:	Analysis
References:	Lesson Plan L1405I, Objective L1405I02RO

Question Number:	SRO 74
Question:	Which of the following describes the operation of the Emergency bus <u>second level</u> undervoltage protection scheme? A. When 1 of 2 relays sense bus voltage less than 95% of nominal for 1.2 seconds (RAT available), it initiates a sequence of load stripping and subsequent bus reenergization by the DG. B. When 1 of 2 relays sense bus voltage drop below 25% of nominal, they initiate auto closure of the RAT supply breaker. C. When both relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), they initiate a sequence of load stripping and subsequent bus reenergization by the DG. D. When both relays sense bus voltage less than 95% of nominal coincident with an SI existing for greater than 10 seconds, they initiate a sequence of load stripping and subsequent bus reenergization by the DG.
Answer:	D
Justification:	A, B, & C are incorrect as none of these answers provide the correct coincidence AND correct setpoint for 2nd level U/V protection. D is correct IAW 1-NHY-310102 sh A53h.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	062K4.01
K/A Values:	3.2
Cognitive Level:	Memory (I)
References:	Detailed Systems Text, 4.16KV Distribution System, pages 21 & 22. Schematic Diagram, 1-NHY-310102 sh A53h. Objective L1093I13RO

QUESTION: 011

Which of the following describes the operation of the Emergency bus first level undervoltage protection scheme?

a. 2 normally energized undervoltage relays. When one of 2 relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), it initiates a sequence of load stripping and subsequent bus reenergization by the DG.

b. 2 normally energized undervoltage relays. When bus voltage drops below 25% of nominal, they deenergize, initiating auto closure of the RAT supply breaker.

c. 2 normally energized undervoltage relays. When both relays sense bus voltage less than 70% of nominal for 1.2 seconds (RAT available), they initiate a sequence of load stripping and subsequent bus reenergization by the DG.

d. 2 normally energized undervoltage relays. When both relays sense bus voltage less than 95% of nominal coincident with an SI existing for greater than 10 seconds, they initiate a sequence of load stripping and subsequent bus reenergization by the DG.

ANSWER:

c.

KA RATING	RO	SRO
062 K4.01		3.2

REFERENCES:

Detailed Systems Text, 4.16KV Distribution System, pages 21 & 22

OBJECTIVES:

L1093I13RO

JUSTIFICATION:

Answer "a" is incorrect because the sequence is initiated by 2 of 2 relaying, not 1 of 2. Answer "b" is incorrect because it describes the 2 relays (of the 6 on the emergency bus) that drop out to provide an auto transfer to the RAT when the UAT is lost or the UAT breaker trips open. Answer "d" is incorrect because it describes the second level undervoltage protection scheme.

Question Number:	SRO 87
Question:	The crew is preparing to perform a rapid plant shutdown from 100% power using Figure 6: Rapid Power Decreases Guidelines, of OS1000.06, POWER DECREASE. The Unit Supervisor directs the PSO to turn on the 'A' and 'B' pressurizer Backup Heaters to force pressurizer spray. What is the reason for this direction? A. Ensures proper mixing occurs such that RCS loop boron samples accurately reflect actual boron concentration. B. Prevents boron stratification in the pressurizer spray nozzles. C. Prevents an RCS dilution event from occurring during a pressurizer outsurge. D. Ensures adequate flow through the pressurizer to prevent boron precipitation on the pressurizer heaters.
Answer:	C
Justification:	A is incorrect because PZR spray operation ensures proper mixing in the PZR and resultant representative PZR boron samples <u>not</u> RCS loop boron samples. B is incorrect because this phenomenon is fictitious but plausible, i.e. sounds similar to RCS loop temperature stratification. C is correct because pressurizer heater operation causes continuous PZR spray which ensures a minimum boron concentration differential between the PZR and the RCS loops precluding an inadvertent boron dilution event as a result of a PZR outsurge of water into the RCS. D is incorrect, boric acid will not precipitate on the PZR heaters as long as the heaters are covered with water.
Direct/New/ Modified	Modified from 1996 NRC exam
K/A #:	2.1.32
K/A Values:	3.4/3.8
Cognitive Level:	Memory (I)
References:	OS1000.07, APPROACH TO CRITICALITY INPO SOER 94-02, Boron Dilution Events Objective L1167I02RO

QUESTION: 087

OS1000.01, "Heatup From Cold Shutdown to Hot Standby", states the following Limitation:

> "When changing RCS boron concentration, pressurizer sprays should be utilized to maintain the differential boron concentration between the pressurizer and reactor coolant loops to less than 50 ppm."

What is the reason for this Limitation?

a. Ensures proper mixing occurs such that RCS loop boron samples accurately reflect actual boron concentration.

b. Prevents boron stratification in the pressurizer spray nozzles.

c. Prevents an RCS dilution event from occurring during a pressurizer outsurge.

d. Ensures that the boron concentration used in the accident analyses remain in their analyzed range of values.

ANSWER:

c

KA RATING	**RO**	**SRO**
A2.1.32	3.4	3.8

REFERENCES:

OS1000.01, Heatup From Cold Shutdown to Hot Standby.
INPO SOER 94-02, Boron Dilution Events.

OBJECTIVES:

L1167I02RO

Answer a. is incorrect because PZR spray operation ensures proper mixing in the PZR and resultant representative PZR boron samples <u>not</u> RCS loop boron samples.

Answer b. is incorrect because this phenomenon is fictitious but plausible, i.e. sounds similar to RCS loop temperature stratification.

Answer c. is correct because pressurizer heater operation causes continuous PZR spray which ensures a minimum boron concentration differential between the PZR and the RCS loops precluding an inadvertent boron dilution event as a result of a PZR outsurge of water into the RCS.

Answer d. is incorrect because this statement reflects the boron concentration of the entire RCS not just the PZR and initial boron concentration of the RCS is not considered in the UFSAR accident analyses.

Seabrook SRO Examination
Work Sheet - Draft

Question Number:	SRO 91
Question:	While work is being performed on Feed water heater drain components, the CONTACT PERSON for one work package requests a temporary lift. There are two additional work packages assigned to the clearance. The TAGGING AUTHORITY is unable to locate the CONTACT PERSON on one work package. How should the request be processed? A. Obtain concurrence from another SRO licensed individual and designate an alternate CONTACT PERSON for notification of the temporary lift, then approve the request after notification is made. B. Do NOT approve the request until the designated CONTACT PERSON is located. C. Identify another Level 1 individual who will assume the CONTACT PERSON responsibilities, then approve the request for temporary lift. D. Temporarily assign the person requesting the temporary lift the CONTACT PERSON responsibility for all work packages under the clearance while the temporary lift is in effect. When responsibility has been assumed, approve the request.
Answer:	C
Justification:	A is incorrect because another SRO licensed individual is not required. B is incorrect because the procedure provides for assigning an alternate CONTACT PERSON. C is correct as outlined in sections 4.6 and 4.9 of MA 4.2. D is incorrect because the new CONTACT PERSON is NOT required to assume the responsibility for all work packages under that clearance.
Direct/New/ Modified	Modified from 1998 NRC exam
K/A #:	2.2.13
K/A Values:	3.8
Cognitive Level:	Memory (I)
References:	MA 4.2, section 4.5 Lesson Plan L1501I, Objective L1501I17RO