INITIAL SUBMITTAL OF THE WRITTEN EXAMINATION

FOR THE PRAIRIE ISLAND INITIAL EXAMINATION

THE WEEKS OF SEPTEMBER 10 AND 17, 2001

Record #	1
RO #	
SRO #	1

Stem:

(see reference)

The following conditions exist on Unit 2:

- A reactor startup is in progress
- Core Exposure is 12,000 MWD/MTU
- The estimated critical position calculated for the startup is 23 steps on Control Bank D
- The latest 1/M plot indicates criticality at 63 steps on Control Bank C
- The control rods indicate the following:
  - The Shutdown Banks and Control Bank A are fully withdrawn
  - Demand position for Control Bank B is at 162 steps
  - Rod Position Indication for Control Bank B rods are 160, 162, 162, 162
  - Demand position for Control Bank C is at 37 steps
  - Rod Position Indication for Control Bank C rods are 36, 40, 38, 16, 38
- Annunciator **47013-0407** ROD AT BOTTOM is lit
- Tavg and Nuclear Instrumentation indications are stable

What action must be taken concerning these conditions?

Answers/Distracters:

a. Fully **reinsert** the control bank rods, AND begin immediate **boration** to hot shutdown boron concentration.

b. Fully **reinsert** the control and shutdown rod banks AND check the ECC.

c. **Insert** Control Bank C to LOWER THAN the 12 steps withdrawn demand position AND replace lift coil for rod G-7.

d. The startup may continue BUT the Control Rod Indication for rod G-7 must be **repaired** before exceeding 2% reactor power.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPK	3	4	1	1	Prairie Island	9/10/2001

Basis for answers:	
a	Incorrect- This action is applied if the reactor becomes critical prior to the Rod Insertion Limit. RIL for the above conditions is 47 steps on Bank C. Criticality has not occurred and is not expected to a point above this value.
b	Incorrect-The condition for administrative limits on critical rod position is - 750 pcm from the calculated critical rod position. The predicted critical position is within the 750 pcm limit for 23 steps on Bank D. (1199 pcm + 750 pcm = 1949 pcm) Corresponding rod position limit is between 42 and 61 steps withdrawn on Control Bank C (closer to 42 steps). 63 steps is well above this value. If it were not, the action is to reinsert rods and reperform the ECC, and adjust as necessary
c	Incorrect- This action may be considered if the rod was misaligned. Two items affect this answer: 1) Indications do not support misaligned rod and 2) the rod misalignment requirement is 24 steps difference between a RCCA and the other RCCAs in the bank.
d	Correct- The indication is that of drift or failure of the control rod position indication for rod G-7. The Rod Bottom alarm for Bank C became active as rod position passed 35 steps withdrawn, and alarmed due to the one rod being less than 20 steps. Entry into 2C5 AOP 4 is specified in the Alarm Response and the operator is directed to 2C5 AOP 5 for the RPI failure. There the operator is directed to implement Tech Spec 3.10.F. The Tech Spec is applicable in MODE 1 (> 2% reactor power) such that with Bank Demand Position between 30 and 215 steps, the difference between individual and group position indication shall be no greater than ± 12 steps.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.1	Conduct of Operations	2.1.7	3.7	4.4
K/A Statement:	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.			

Reference Title:	Reference #:	Section:	Page:	Rev:
Tech Spec Intro/Overview	P8171L-007	III.D.2	16	2
MISALIGNED ROD, STUCK ROD, AND/OR RPI FAILURE OR DRIFT	2C5 AOP5	2.5.1, 2.5.2	7	3
Prairie Island Technical Specifications		3.10.F	TS.3.10-6A	139

Facility Learning Objective:	P8171L-007 #4
Question Source:	New
Comments:	Material Required for Examination Fig C1-4B. (Rod Worth)

Record #	2
RO #	
SRO #	2

Stem:

Given the following:

The site tour coordinator has contacted the Unit 1 Shift Supervisor and asks if there are any extra operators available to act as escorts during a planned site tour for a group of 17 people.

The tour schedule is as follows:

- 10:00 AM – Tour of turbine floor
- 11:00 AM – Tour of Spent Fuel Pool

What is the minimum number of escorts required for each leg of the tour?

Answers/Distracters:

a. 4 escorts for the Turbine Floor tour
4 escorts for the Spent Fuel Pool tour

b. 2 escorts for the Turbine Floor tour
4 escorts for the Spent Fuel Pool tour

c. 2 escorts for the Turbine Floor tour
2 escorts for the Spent Fuel Pool tour

d. 4 escorts for the Turbine Floor tour
2 escorts for the Spent Fuel Pool tour

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-F	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-This could be selected if the candidate thinks that the maximum visitor to escort ratio is 5:1 at any time.
b	Correct-Escorts should normally escort (1) visitor at a time in work related situations. Exemptions may be granted by the Security Supervisor for meetings and authorized tours. The maximum visitor to escort ratio is 10:1 in protected areas and 5:1 in vital areas. Since the turbine floor is a protected area 2 escorts are required for the 17 visitors. The Spent Fuel Pool is a vital area therefore it requires 4 escorts for the 17 visitors.
c	Incorrect- This could be selected if the candidate thinks that the maximum visitor to escort ratio is 10:1 at any time.
d	Incorrect-This could be selected if the candidate doesn't understand the protected and vital area boundaries.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
2.1 Conduct of Operations	2.1.13		2.9

K/A Statement: Knowledge of facility requirements for controlling vital/controlled access

Reference Title:	Reference #:	Section:	Page:	Rev:
Plant Access	P7720C-001	III.C.4	35-36	0
Security Policies and Procedures	5AWI 5.1.1	6.3.5	7	14

Facility Learning Objective:	P7720C-001 part III #9
Question Source:	new
Comments:	

Record #	3
RO #	1
SRO #	3

Stem:

Given the following conditions on Unit 1:

- On the previous evening the operators recorded the incorrect number for the Boric Acid Tank placed on recirculation during that shift.
- Tonight, halfway through the shift, the operators realize that they had made the error concerning the Boric Acid Tank.

How do the operators make the required corrections?

Answers/Distracters:

a. The correct number is inserted directly into the log for the previous night shift AND automatically prints the corrected legal copy of the log.

b. The correct number is inserted directly into the log for the previous night shift, but the legal copy of the log for that day will NOT reflect the change.

c. A corrected entry is made in the current shift log referring to the date and time of the incorrect log entry AND is printed out on the next legal copy of the log.

d. A corrected entry is made in the current shift log that refers to the date and time of the incorrect log entry AND a handwritten change is made to the incorrect entry on the printed legal copy of the log.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-RW	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-The correction can be made directly to the log if it had not been archived. Automatic printing of the legal copy does not occur.
b	Incorrect-This would be possible if the log had not been archived that morning (as directed by SWI O-25). The legal copy does not contain the corrected information if already printed. This is noted in SWI O-25 for late entry additions to the log.
c	Correct-Errors in log entries can be edited at any time prior to archiving. If an entry is in error and has already been archived, the corrected entry SHALL refer to the date and time of the entry that was in error. At the beginning of the day shift, the Unit 1 SS SHALL print the Operations log after taking the duty. The printed log will go into the "OUT" basket for routing, and after the log is printed the SS will archive the Operations Log. (The printed log is a legal copy that needs to be kept for the life of the plant.) Since it is the night shift, the previous days log has already been archived.
d	Incorrect-No change is made to the legal copy. The next days legal copy will reflect the corrected information.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.1	Conduct of Operations	2.1.18	2.9	3.0

K/A Statement: Ability to make accurate, clear, and concise logs, records, status boards, and reports.

Reference Title:	Reference #:	Section:	Page:	Rev:
Records/Logs	P9150L-012	VI.B	13	2
PERIODIC DATA ACQUISTION & LOGKEEPING	SWI O-25	6.3.2, 7.2.6	5,7	26

Facility Learning Objective:	P9150L-012 #6.b.i
Question Source:	New
Comments:	

Record #	4
RO #	2
SRO #	

Stem:

Under which of the following conditions would physical Independent Verification of a SI pump discharge valve NOT be required?

Answers/Distracters:

a. The valve is being closed per an isolation step of a work order that places safety tags on the suction and recirc valves during POWER OPERATION.

b. The valve is being RESTORED during an outage AND the SI system checklist is to be performed prior to leaving COLD SHUTDOWN.

c. The SI system checklist is being performed prior to leaving COLD SHUTDOWN AND the first checker finds the valve in the OPEN position.

d. The valve is being OPENED per an I&R during HOT SHUTDOWN AND entry into the SI Pump Area requires double Anti-C's and respirator use due to contamination levels.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- IV is still required if the valve is positioned from its normal operating position even if it is not part of the boundary tagout
b	Correct- Two conditions identified when physical IV are NOT required: 1) a. A significant amount of radiation exposure would be received (ALARA reasons). OR 2) b. For outage related safety tag restorations if a system checklist is to be completed prior to the required operability for each component involved.
c	Incorrect- IV is still required during checklist performance even if the first operator finds the valve in its proper position.
d	Incorrect- IV is required unless the radiation levels are deemed excessive, not the protection requirements required by the operator.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.1	Conduct of Operations	2.1.29	3.4	3.3

K/A Statement:	Knowledge of how to conduct and verify valve lineups.

Reference Title:	Reference #:	Section:	Page:	Rev:
Control/Operation of Plant Equipment	P9150L-024	V.D.1	20-21	0
METHODS OF PERFORMING INDEPENDENT VERIFICATION	5AWI 3.10.1	6.0, 6.1	5-8	8

Facility Learning Objective:	P9150L-024 #4
Question Source:	Facility Exam Bank
Comments:	Q # P9150L-024 025. Minor wording changes to 2 selections

Record #	**5**
RO #	**3**
SRO #	**4**

Stem:

The following conditions exist on each Unit:

- The control room has been evacuated due to a hostile environment
- Unit 1 was at 50% power and Unit 2 was in COLD SHUTDOWN when the control room was evacuated
- Unit 1 operators were able to perform all the listed control room operations of 1C1.3 AOP1 "SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 1 prior to exiting
- S/G levels are currently at 12% and 16% narrow range for 11 and 12 S/Gs, respectively
- The operators are taking the actions to control S/G levels per 1C1.3 AOP1

Which of the following describe the local actions the operators will take to control the AFW Pumps?

Answers/Distracters:

a. **Place** ONLY 11 TD AFW control to "LOCAL", AND **check** the TD AFW pump continues to run.

b. **Place** ONLY 12 MD AFW control to "LOCAL", AND **press** the local start pushbutton to restart the MD AFW pump.

c. **Place** BOTH 11 TD AFW and 12 MD AFW Pump controls to "LOCAL", AND **check** both pumps continue to run.

d. **Place** BOTH 11 TD AFW and 12 MD AFW Pump controls to "LOCAL", **check** the TD AFW pump continues to run AND **press** the local start pushbutton to restart the MD AFW pump.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
c	2-DR	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This may be the action required if only the Turbine Driven pump were running, but due to S/G level both pumps should be currently running
b	Incorrect- This may be the action required if only the Motor Driven pump were running, but due to S/G level both pumps should be currently running. Also, a local restart of the pump is NOT required
c	Correct- Both pumps auto started (2/3 S/G level < 13%) and should be running. Both pumps controls should be placed in LOCAL. When this action is performed, the pump operation should not be affected. This will occur whether the S/G level is above or below the automatic start setpoint of 13%
d	Incorrect-- Local manual restart of the pump is NOT required when the pump controls are taken to LOCAL.

K/A System/Evolution:		*K/A #:*	*KAVRO*	*KAVSRO*
2.1	Conduct of Operations	2.1.30	3.9	3.4

K/A Statement: Ability to locate and operate components, including local controls.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
1C1.3 AOP1 Review	P8197L-008	II.F.8	11	4
Auxiliary Feedwater System	P8180L-007	IV.A.2, B.1	18, 21-22	5
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 1	1C1.3 AOP1	2.4.15	7	4

Facility Learning Objective:	P8197L-008 #3 P8180L-007 #3,4
Question Source:	New
Comments:	

Record #	6
RO #	4
SRO #	5

Stem:

During a Control Room evacuation, what are the designated Sound Power Communications channels used by plant personnel as directed in 1C1.3 AOP1 "SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 1" and 2C1.3 AOP1 "SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 2"?

Answers/Distracters:

a. Unit 1 uses Channel 1 preferred and Channel 2 alternate, while Unit 2 uses Channel 3 preferred and Channel 4 alternate.

b. Unit 1 uses Channel 1 preferred and Channel 3 alternate, while Unit 2 uses Channel 2 preferred and Channel 4 alternate.

c. Unit 1 and Unit 2 use Channel 1 preferred while Unit 1 Uses Channel 2 alternate and Unit 2 uses Channel 3 alternate.

d. Unit 1 and Unit 2 use Channel 1 preferred and Channel 2 alternate.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- (see b. below)
b	Correct- Procedural designation for 1C1.3 AOP 1: "All Operations personnel will establish sound powered phone communications. Channel 1 - preferred, Channel 3 - alternate.", and 2C1.3 AOP 1: "All Operations personnel will establish sound powered phone communications. Channel 2 - preferred, Channel 4 - alternate." Other selections represent other possible selections available to operators.
c	Incorrect- (see b. above)
d	Incorrect- (see b. above)

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
2.2 Equipment Control	2.2.4	2.8	3.0

K/A Statement:	(multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.

Reference Title:	Reference #:	Section:	Page:	Rev:
1C1.3 AOP1 Review	P8197L-008	II.F.5.e	10	4
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 1	1C1.3 AOP1	2.4.8 NOTE	5	4
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 2	2C1.3 AOP1	2.4.8 NOTE	5	5

Facility Learning Objective:	P8197L-008 #5
Question Source:	New
Comments:	

Record #	7
RO #	
SRO #	6

Stem:

The following conditions exist:

- Operators are performing an approved procedure on unit 2.
- Step 23 of the procedure, has a temporary change notice, TCN, signed by the Unit 1 SS (Reviewer) and Unit 2 SS (Approver) requiring Valve "A" to be throttled OPEN to obtain a flow rate of 200 gpm.
- The design limit specified in the Precautions and Limitations Section is 175 gpm.

Which of the following actions must be taken with regard to performing step 23?

Answers/Distracters:

a. Do NOT perform the step. TCNs that change the intent or scope are NOT permitted unless signed by the OC or assigned approver.

b. **Perform** the step as written AND **submit** a TCN to modify the Precautions and Limitations Section.

c. **Perform** the step as written AND have the Shift Supervisor **initial** the step.

d. Do NOT perform the step UNTIL TWO Licensed Senior Reactor Operators view AND approve another TCN.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-DR	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- Temporary changes that change the intent of the procedure require additional reviews before use. The following are changes in intent: a) A change to the purpose of the procedure; b) Performing activities in addition to, or instead of, what is the overall purpose of the procedure; c) Changes to acceptance criteria; d) Changes to tolerances that affect equipment operability; e) Changes involving a commitment. The TCN has the operator run the system outside of design limits which is a change in intent and requires OC approval.
b	Incorrect- The TCN must be approved by the OC prior the implementing the step.
c	Incorrect- Only the OC can approve a Change in intent.
d	Incorrect-Temporary changes that change the intent of the procedure require additional reviews before use. Temporary changes to OC reviewed procedures/critical work orders require the concurrence (Originator, Reviewer, or Approver) of two (2) members of unit management staff, at least one of whom holds an SRO license. This change must be approved by the OC.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.2	Equipment Control	2.2.8	1.8	3.3

K/A Statement:	Knowledge of the process for determining if the proposed change, test, or experiment involves an unreviewed safety question.

Reference Title:	Reference #:	Section:	Page:	Rev:
Operability and TS Essential Equipment Database	P8171L-009	II.C.6.a	17	0
Procedures	P9150L-003	IV D,E	26-27	2
SAFETY EVALUATION SCREENINGS	5AWI 3.3.3	6.2	8-9	19

Facility Learning Objective:	P9150L-003 #11.a
Question Source:	Facility exam bank question P9150L-003 022
Comments:	Minor wording changes

Record #	**8**
RO #	**5**
SRO #	**7**

Stem:

A Quarterly Surveillance test has a due date of June 15th. What is the latest date this test could be performed and still be within the required test interval time? **(June has 30 days)**

Answers/Distracters:

a. June 22nd

b. July 6th

c. July 15th

d. July 28th

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
b	3-SPK	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-This is the time for a bi-monthly test...±7 days
b	Correct-This is the time interval for a quarterly test. ±21 days which represents ±25% of the surveillance period from the due date..
c	Incorrect-This would be the correct time period if 1 month were thought to be 25% of a quarterly time period.
d	Incorrect-This is the time period for a semi-annual test.. ±45 days.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
2.2 Equipment Control	2.2.12	3.0	3.4

K/A Statement: Knowledge of surveillance procedures.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Surveillance Program	P9150L-016	IV.F.1.e	14	1
SURVEILLANCE AND PERIODIC TEST PROGRAM	G1	2.12	6	19

Facility Learning Objective:	P9150L-016 #4
Question Source:	new
Comments:	

Record #	9
RO #	6
SRO #	8

Stem:

The following conditions exist on Unit 1:

- Reactor power is 20% during a power increase.
- 11 CC Pump is out of service for motor repair.
- 11 RHR Heat Exchanger is isolated on RHR side to facilitate testing for a possible tube leak.
- 12 Charging Pump is out of service.

Which of the following, if found to be inoperable, would require that a plant shutdown for Unit 1 be commenced within ONE hour as directed by Technical Specifications?

Answers/Distracters:

a. D1 Diesel Generator

b. 11 Charging Pump

c. 12 SI Pump

d. 13 CFCU

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-RI	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-) D1 may be inoperable as long as the other DG (D2) is demonstrated operable and the same train ESF equipment is operable. The required Unit 1 equipment is operable.
b	Incorrect- Prior to a recent revision to the facility's Tech Specs, 2 of 3 charging pumps inoperable was a 72 hour LCO.
c	Correct-During STARTUP and POWER OPERATIONS ONE of the following may be inoperable within the proper time limits (72 hours) - One SI system and One RHR System, provided the redundant SI system and RHR system required for functioning during accident conditions is operable. The RHR HX and proposed SI Pump are on opposite trains; and therefore, TS 3.0.C must be implemented since the TS cannot be met
d	Incorrect- The CFCU is a Train A component and is only affected if Containment Spray pump is inoperable.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
2.2 Equipment Control	2.2.24	2.6	3.8

K/A Statement: Ability to analyze the affect of maintenance activities on LCO status.

Reference Title:	Reference #:	Section:	Page:	Rev:
Tech Spec Intro/Overview	P8171L-007	III.D.2	16	2
Prairie Island Technical Specifications		3.3.A.2	TS 3.3-2	91
Prairie Island Technical Specifications		3.0.C	TS 3.0-1	91

Facility Learning Objective:	P8171L-007 #2,4
Question Source:	Facility Exam Bank
Comments:	P8171L-007 002. Changed conditions from 100% power. Changed 2 of the components out of service. Changed two selections: 1) from Unit 1 DG1 to Unit 2 DG6 and 2) from Unit 1 Cont. Spray Pump to CFCU.

Record #	10
RO #	
SRO #	9

Stem:

(see reference)

The following conditions exist on Unit 2:

- Refueling operations are in progress
- Refueling cavity boron concentration was measured at 1950 ppm
- The calculated Keff is 0.94

What action is required to be taken concerning fuel movement?

Answers/Distracters:

a. Fuel movement May CONTINUE since NO reactivity conditions are violated.

b. Fuel movement May CONTINUE BUT boration must be initiated to restore required boron concentration.

c. Fuel movement Shall CEASE AND boration must be initiated to restore the required shutdown margin (lower Keff).

d. Fuel movement Shall CEASE AND boration must be initiated to restore the required boron concentration.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-P	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Boron concentration is lower than allowed by TS Table 1-1
b	Incorrect- Action taken to increase boron concentration is correct. Core Alteration may NOT continue and actions resulting in change in core reactivity is allowed, provided it does NOT result in an increase in reactivity (actions resulting in decrease in core reactivity may occur).
c	Incorrect- The SDM and Keff requirements are met and no direction exist for the limitation of double the SR counts.
d	Correct- Technical Specification 3.8-2 requires the plant be in REFUELING condition (MODE 6) to allow CORE ALTERATIONS. Table TS 1-1 identifies the reactivity condition required as to ensure the most restrictive condition of 3 be met: 1) Keff </= 0.95; 2) Boron concentration >/= 2000 ppm; or 3) SDM as specified in COLR. The given information indicates Keff and SDM requirements are met but boron concentration is lower than minimum required. TS 3.8.A.2 indicates if the condition is NOT met, Core Alterations must be stopped, action taken to correct the violated condition, and no operations which may increase the reactivity of the core shall be performed.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.2	Equipment Control	2.2.28	2.6	3.5

K/A Statement: Knowledge of new and spent fuel movement procedures.

Reference Title:	Reference #:	Section:	Page:	Rev:
Fuel Handling	P8182L-003	V.B.2.a,b	68-69	4
Prairie Island Technical Specifications		3.8.A	TS 3.8-1, 2	130/119
Reactor Refueling Operations	D5.2	3.0	4	27
Prairie Island Technical Specifications		Table TS. 1-1	TS 1-1	156

Facility Learning Objective:	P8182L-003 #18
Question Source:	Facility Exam Bank
Comments:	Requal Part B, P9140L-603 002. Modified initial conditions, changing the affected parameter. Changed 3 selections to include identification of the affected parameter (use of required boration curve), and changed correct answer.

Record #	11
RO #	7
SRO #	10

Stem:

A 21 year old male licensed operator with a June birthday has the following exposure history:

- A Committed Effective Dose Equivalent (CEDE) of 0 mRem
- A Current YEARLY whole body Deep Dose Equivalent (DDE) of 625 mRem
- A LIFETIME Total Effective Dose Equivalent (TEDE) of 5 Rem

Assuming his exposure is properly documented. What is the maximum ADDITIONAL whole body exposure this operator can receive this Year and still comply with the Prairie Island administrative limit?

Answers/Distracters:

a. 375 mRem

b. 1375 mRem

c. 2375 mRem

d. 4375 mRem

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	3-SPK	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:	
a	Incorrect- This could be considered if the Prairie island administrative limits were thought to be set at 20% of the federal limits for TEDE.
b	Correct-Prairie Island sets administrative dose limits at 40% of the 10 CFR 20 limits of 5000 mRem/year. The TEDE is equal to The DDE+CEDE. In this case the operator has a TEDE of 625 mRem for the year. Under the Prairie Island administrative limits he could receive 2000-625=1375 mRem for the rest of the year.
c	Incorrect-This could be considered if the Prairie island administrative limits were thought to be set at 60% of the federal limits for TEDE.
d	Incorrect-This could be considered if the Prairie Island limits were thought to be the same as the federal limits.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.3	Radiation Control	2.3.1	2.6	3.0

K/A Statement:	Knowledge of 10 CFR: 20 and related facility radiation control requirements.

Reference Title:	Reference #:	Section:	Page:	Rev:
F2 Radiation Safety	P9130L-003	V.A	11	2
RADIATION SAFETY	F2	4.2.3	22	20

Facility Learning Objective:	P9130L-003 #3
Question Source:	New
Comments:	

Record #	12
RO #	8
SRO #	

Stem:

In accordance with the ALARA Program, which of the following describes an action taken at Prairie Island used to minimize the annual integrated dose for all workers?

Answers/Distracters:

a. All Hot Spots are shielded with Portable shielding.

b. Dissolved hydrogen is maintained in the RCS during power operations.

c. The CVCS letdown flow rate is maintained at its minimum value during plant outages.

d. Power changes are performed at the maximum permissible rate as allowed by procedure.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-F	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- An ALARA evaluation is performed for each case to determine if shield installation provides a relative dose reduction. Installation/removal of the shielding in some cases may result in overall increase in exposure (MAN-REM) for a task.
b	Correct- In order to minimize the formation of corrosion products and fission products and methods used to clean up the products if they are released to the RCS, the Reactor Coolant System is operated with dissolved hydrogen in the water which reduces the corrosion rates of the metal in the system. (Among other actions.)
c	Incorrect-- Purification of the RCS is maximized during outages, shutdowns, and startups to reduce radiation levels.
d	Incorrect- The power rate changes are minimized to help reduce the amount of fission products that are released to the RCS.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.3	Radiation Control	2.3.2	2.5	2.9

K/A Statement:	Knowledge of facility ALARA program.

Reference Title:	Reference #:	Section:	Page:	Rev:
F2 Radiation Safety	P9130L-003	I.B.1	8	2
RADIATION SAFETY	F2	1.3.1	4	20

Facility Learning Objective:	P9130L-003 #1
Question Source:	Facility Exam Bank
Comments:	P9130L-003 028. Minor wording changes (shorten question)

Record #	13
RO #	
SRO #	11

Stem:

Given the following conditions:

- A LOCA outside containment has occurred
- The Shift Manager has assumed the role of the Emergency Director
- The faulted line was manually isolated locally, however the operator performing the task was injured and CANNOT leave the area on his own
- Initial dose rate estimates for the area are 75 R/hr
- The recovery time for the injured operator using a search and rescue team is estimated to take 10 minutes.

Which of the following describes the conditions concerning a rescue attempt?

Answers/Distracters:

a. NO attempted rescue may be made since the exposure will exceed the allowed dose guidelines.

b. NO special authorization is required since this exposure will NOT exceed 10CFR20 NRC limits.

c. Only qualified individuals, selected by the Emergency Director, may attempt the rescue WITH the approval of the Emergency Director.

d. Only volunteers, after being made aware of all risks, can attempt the rescue WHEN authorized by the Emergency Director.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-SPK	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- There is NO specific exposure limit on life-saving efforts in a declared emergency.
b	Incorrect-10CFR20 limits are exceeded by this action.
c	Correct- In the event of a planned exposure in excess of the 10CFR20 NRC limits, the following procedure SHALL be followed. If necessary, the Emergency Director may verbally authorize increased exposure when time is a limiting factor and documentation SHALL be completed as a follow-up. The Emergency Director (initially the Shift Manager) shall authorize all exposure in excess of NRC 10CFR20 limits (5 Rem TEDE). Table 1 provides the Criteria for Emergency Exposure: Life saving or protection of large populations (operate vital equipment) has a 25 Rem TEDE limit. This limit may be exceeded only on a voluntary basis to persons fully aware of the risks involved. The projected exposure in this rescue is 75 REM/HR x 0.167 = 12.5 REM, with a maximum expected exposure of 18.75 Rem (15 minutes). This is above the 10CFR20 limits but below the level requiring voluntary action.
d	Incorrect- the requirement for a volunteer is only specified if the exposure is expected to exceed 25 REM TEDE. It is NOT expected to exceed this.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.3	Radiation Control	2.3.4	2.5	3.1

K/A Statement:	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Reference Title:	Reference #:	Section:	Page:	Rev:
EMERGENCY EXPOSURE CONTROL	P7420L-003	II.B.3, 4	9	3
EMERGENCY EXPOSURE CONTROL	F3-12	7.2.1, 8.0, Table 1	3,4,12	14
ONSITE EMERGENCY ORGANIZATION	F3-1	4.1.1	5-7	19

Facility Learning Objective:	P7420L-003 #2,3
Question Source:	New
Comments:	

Record #	14
RO #	
SRO #	12

Stem:

The following conditions exist on Unit 1:

- The plant is at 100% power
- Flux mapping is in progress
- The Personnel Airlock inner air lock has FAILED its leak rate test AND maintenance is awaiting parts for repair
- THREE personnel are planning to enter containment to take local vibration readings on 13 CFCU

Under these conditions when is Containment entry allowed?

Answers/Distracters:

a. Containment entry can be made only when continuous RPS monitoring of radiation levels at 13 CFCU is available.

b. Containment entry can be made only when flux mapping is completed.

c. Containment entry can be made only when reactor power is reduced to less than 50%.

d. Containment entry can be made only when the Personnel Airlock door is repaired AND tested.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- Monitoring is required during at-power entries but the flux mapping requirement is NOT met.
b	Correct- Prior to containment entry, the Shift Supervisor is to confirm there is no flux mapping or incore detector movement in progress. The incore detectors can cause Very high radiation dose rates and possible overexposures.
c	Incorrect- There is no power limit on at-power entries.
d	Incorrect- When all personnel are out of the containment the personnel and maintenance airlock hatches shall be locked, but no requirement exists for both to be operable for entries. Personnel will use the Maintenance Airlock.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.3	Radiation Control	2.3.10	2.9	3.3

K/A Statement:	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Reference Title:	Reference #:	Section:	Page:	Rev:
F2 Radiation Safety	P9130L-003	VI.B.3	12-13	2
RADIATION SAFETY	F2	9.2.3	30	20

Facility Learning Objective:	P9130L-003 #5
Question Source:	Facility Exam Bank
Comments:	Slightly modified Reworded exam bank question P9130L-003 007 SRO Only.

Record #	15
RO #	9
SRO #	

Stem:

The following conditions exist:

- 123 Gas Decay Tank (GDT) release is in progress
- Releases for 124 and 125 GDT have been approved and are waiting release

What action is to be taken if the Met Tower data link fails during the 123 GDT release?

Answers/Distracters:

a. When the 123 GDT release is completed, do NOT initiate another GDT release UNTIL Met Tower data is available.

b. **Direct** the Auxiliary Building operator to STOP the release of 123 GDT, AND do NOT initiate another GDT release until Met Tower data is available.

c. **Contact** Lock and Dam #3 periodically for wind conditions while CONTINUING the release for all GDTs.

d. **Contact** the National Weather Service periodically for wind conditions while continuing the release for all GDTs.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-F	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-) Any release shall be stopped if the limitations cannot be verified. Additionally it is correct the approval of additional releases my not occur until the met tower data is restored.
b	Correct- Authorization for release and during release periodic monitoring of ERCS group display "OPWIND" for 10 meter wind speed and wind direction is required. The release SHALL be terminated if the conditions in Section 3.0 are not satisfied. Loss of the link prevents monitoring these limitations.
c	Incorrect- Local wind conditions and speeds must be available during releases.
d	Incorrect- Local wind conditions and speeds must be available during releases.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.3	Radiation Control	2.3.11	2.7	3.2

K/A Statement: Ability to control radiation releases.

Reference Title:	Reference #:	Section:	Page:	Rev:
Rad Waste - Waste Gas	P8182L-001C	VIII.C	25	2
RELEASING RADIOACTIVE GAS FROM 123 LOW LEVEL GAS DECAY TANK	C21.3-10.3	7.14, 3.0	9,3	12W

Facility Learning Objective:	P8182L-001C #8
Question Source:	Facility Exam bank
Comments:	P8182L-001C 003 Changed layout of premise only.

Record #	16
RO #	10
SRO #	13

Stem:

Which of the following would prevent Containment In-Service Purge from being placed in operation on Unit 1?

Answers/Distracters:

a. Containment Pressure gauge **4127901** is reading 0.47 psig.

b. Annunciator **47021-0301**, CVI TRAIN A DC FAILURE, is lit.

c. The **1R11/12** Sample Selector Switch is in the "VENT" position.

d. The **1R-22** setpoint is below the calculated setpoint listed in the Containment Pre-release Authorization.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	2-RI	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Containment pressure is required to be less than 0.5 psig due to the design specifications of the in-serv purge ducting. With pressure at 0.47 psig this requirement is met.
b	Correct-Both safeguard racks are required to be energized. This is verified by observing that the power supply annunciators are not lit. With annunciator 47021-0301 lit the containment in-service purge can not be placed in operation.
c	Incorrect- With the R11/12 Sample Selector Switch in the "vent" position R11/12 will be sampling the shield bldg. vent stack. This provides for monitoring of the effluent path of the purge. R11/12 is required to be placed in the "vent" position prior to placing in-serv purge in operation. This answer could be considered if the operator thinks that the containment atmosphere must be sampled instead of the shield bldg. Stack as in the case of the Containment Purge.
d	Incorrect-The 1R-22 setpoint is required to be at or below the calculated setpoints in the pre-release authorization form. This answer could be considered if the operator thinks that the setpoint needs to be higher than the calculated setpoint to prevent an auto vent isolation from high rad on 1R-22.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
2.3 Radiation Control	2.3.9	2.5	3.4

K/A Statement: Knowledge of the process for performing a Containment Purge

Reference Title:	Reference #:	Section:	Page:	Rev:
Containment Purge & In-Service Purge Ventilation System	P8180L-009E	IV.B.2	17	1
Containment System Ventilation	1C19.2	5.3	15	6

Facility Learning Objective:	P8180L-009E #6
Question Source:	New
Comments:	

Record #	**17**
RO #	**11**
SRO #	**14**

Stem:

The following conditions exist on unit 1:

- Reactor Protection Logic Testing is being performed causing numerous repetitive alarms

What is the correct response concerning the alarms?

Answers/Distracters:

a. The operator may use a pre-job brief in place of individual alarm notifications after the first notification.

b. The operator shall reference the alarm response procedures for ALL alarms received.

c. The operator may prioritize alarms NOT associated with the logic testing AND announce these alarms only if operationally significant.

d. The operator does NOT need to treat the alarms as valid until proven valid.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a	1-P	3	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct-It is acceptable to use a pre-job brief in place of individual notifications after the first notification with SS approval. Use of devices to identify the expected annunciator window is also acceptable.
b	Incorrect- ARPs are required for all unexpected alarms. The alarms received during testing shall be considered expected.
c	Incorrect-Prioritizing and announcing only operationally significant alarms is only appropriate during accident or transient conditions.
d	Incorrect- All alarms shall be treated as valid until proven otherwise.

K/A System/Evolution:		*K/A #:*	*KAVRO*	*KAVSRO*
2.4	Emergency Procedures / Plan	2.4.10	3.0	3.1

K/A Statement: Knowledge of annunciator response procedures.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Plant Operation	P9150L-014	II.C.3	9	2
Conduct of OPs	SWI O-0	ATT 1	8,9	0

Facility Learning Objective:	P9150L-014 #1.f
Question Source:	New
Comments:	

Record #	18
RO #	12
SRO #	15

Stem:

The following conditions exist on Unit 1:

- The unit is in HOT SHUTDOWN during a normal cooldown
- RCS temperature is at 520°F
- Pressurizer pressure is 1700 psig
- At this point, all Unit 1 4.16KV busses lose power (Loss of all AC power)

How would emergency procedure 1ECA-0.0 "Loss of Safeguards AC Power" be used in this situation?

Answers/Distracters:

a. ENTER 1ECA-0.0 immediately upon verification of loss of power to buses 15 and 16.

b. ENTER 1ECA-0.0 ONLY if RCS temperature rises above 540°F.

c. ENTER 1ECA-0.0 ONLY if a safety injection signal occurs also.

d. ENTER 1ECA-0.0 ONLY if power is NOT restored to EITHER bus 15 or bus 16 when RCS temperature reaches 350°F.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-P	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- At one time ECA-0.0 could be entered directly upon identification of loss of power to both Safeguards buses. This is no longer true.
b	Incorrect- RCS temperature is NOT a entry condition for either ECA-0.0 or E-0.
c	Correct- ECA-0.0 is only entered from Step 3 RNO when it is checked at least one Safeguards bus energized. The entry into E-0 occurs only on a reactor trip or a Safety Injection (or if conditions warrant one of these signals).
d	Incorrect- This condition is indicative of the point where RHR may be placed in service and does provide for another potential entry condition to the emergency procedures, particularly E-4"Core Cooling Following Loss Of RHR Flow". This does not direct the operators to ECA-0.0 however.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
2.4	Emergency Procedures / Plan	2.4.14	3.0	3.9

K/A Statement: Knowledge of general guidelines for EOP flowchart use.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	V.B.1	22	2
Loss of Safeguards AC Power	1ECA-0.0	B.1	2	14
Reactor Trip Or Safety Injection	1E-0	B.1, Step 3	2, 4	19

Facility Learning Objective:	P8197L-011 # A1, 3; F20
Question Source:	New
Comments:	

Record #	19
RO #	
SRO #	16

Stem:

During the performance of 1E-1, LOSS OF REACTOR OR SECONDARY COOLANT, the STA notifies the SS of two Red Path conditions-one in Heat Sink the other in Integrity.

Which of the following describes the procedure implementation hierarchy associated with this condition?

Answers/Distracters:

a. **Remain** in 1E-1 until directed to transition to another E-series procedure then address the Red Path conditions.

b. **Transition** to 1FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

c. **Transition** to 1FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS.

d. **Transition** to 1ES-0.0, REDIAGNOSIS to determine which FR procedure to implement.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- After Critical Safety Function Status Tree Scanning begins at step 23 of E-0 or when transition is made out of E-0 to another E-series procedure any Red or Orange condition must be immediately addressed by transition to the appropriate FR procedure. It is inappropriate to remain in E-1 when an unaddressed Red Path condition exists.
b	Correct-Critical safety function status tree scanning is required when directed in 1E-0 step 23 or whenever a transition is made out of 1E-0 to another E-series procedure. Any Red or Orange condition identified requires immediate transition to the appropriate FR procedure. The highest priority Red Path must be entered first then lower priority Red Paths then Orange Paths. In this case the highest priority Red Path would be Heat Sink and require transition to FR-H.1.
c	Incorrect- The highest priority Red Path must be addressed first. Since Heat Sink is of higher priority than Integrity it must be addressed first. This selection might be considered if the Status tree priority is not known.
d	Incorrect-Although 1ES-0.0 Rediagnosis can be entered on operator judgement it does not apply to FR procedures and would be of no help in determining which FR procedure to implement.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
2.4 Emergency Procedures/Plan	2.4.16	3.0	4.0

K/A Statement: Knowledge of EOP implementation hierarchy and coordination with other support procedures.

Reference Title:	Reference #:	Section:	Page:	Rev:
EOP intro-Procedure Review	P8197L-010	III.E.1.a & E.2.e.1	18-20	2
Background information for Critical Safety Function Status Trees	1F-0		1-2	10

Facility Learning Objective:	P8197L-010 #4.a,c,&d
Question Source:	New
Comments:	

Record #	**20**
RO #	**13**
SRO #	**17**

Stem:

Given the following conditions for Unit 2:

- The unit was at 100% power when a reactor trip occurred
- The operator is performing the immediate action steps of 2E-0 "Reactor Trip Or Safety Injection"
- The operator reports safeguards bus 25 ONLY is deenergized

What is the proper action to take?

Answers/Distracters:

a. **Initiate** action to restore power to Bus 25 per 2C20.5 AOP1 "REENERGIZING 4.16 KV BUS 25".

b. **Continue** immediate actions of 2E-0 for SI Actuation.

c. **Transition** to 2ECA-0.0 "Loss Of All Safeguards AC Power".

d. **Place** the feed breaker controls in MANUAL **AND place** the affected components controls in PULL OUT.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a	1-P	3	2	1	1	Prairie Island	9/10/2001

Basis for answers:	
a	Correct- RNO step for "NO" response to "Safeguards buses - BOTH ENERGIZED" states - Initiate action to restore power to deenergized safeguard bus per 2C20.5 AOP1 or 2C20.5 AOP2. There fore the correct operator action is to initiate 2C20.5 AOP1 -- the proceed to the next step of E-0.
b	Incorrect-- Continuing with next action of E-0 is NOT appropriate if action of 2C20.5 AOP1 has NOT been initiated (or at least addressed)
c	Incorrect- Transition to 2ECA-0.0 will occur only if both Safeguards buses are not available.
d	Incorrect-- Action is specific to 2 C20.5 AOP1 after the procedure is initiated and power is ready to be restored to Bus 25. This action is also specified in ECS-0.0 when preparing to reenergize a Safeguards Bus.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
2.4 Emergency Procedures / Plan	2.4.49	4.0	4.0

K/A Statement:	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
EOP Intro-Procedure Review	P8197L-010	III.B	15	2
Reactor Trip Or Safety Injection	2E-0	Step 3	4	18

Facility Learning Objective:	P8197L-010 #2,7
Question Source:	Significantly modified from Facility Exam Bank
Comments:	P8197L-010 005. Change premise conditions from loss of All AC to loss of one bus. changed one answer for actions to be taken. Correct answer changed.

Record #	21
RO #	14
SRO #	

Stem:

Which of the following is **NOT** a basis for maintaining control rods above the Rod insertion limits.

Answers/Distracters:

a. Assures negative reactivity is inserted within 1.8 sec from a reactor trip.

b. Assures adequate trip reactivity.

c. Assures meeting power distribution limits.

d. Limits the consequences of a hypothetical rod ejection accident.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-B	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - This is NOT a basis for maintaining rods above the RIL. 1.8 sec is the maximum rod drop time allowed by Prairie Island Technical Specifications.
b	Incorrect - Adequate trip reactivity is one of the three bases listed for RILs.
c	Incorrect - Meeting power distribution limits is one of the three bases listed for RILs.
d	Incorrect - Limiting the consequences of a hypothetical rod ejection accident is one of the three bases listed for RILs.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
001 Control Rod Drive System	2.2.25	2.5	3.7

K/A Statement: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Reference Title:	Reference #:	Section:	Page:	Rev:
Control Rod Worth	P8188L-016	V.B	14-15	1
Prairie Island Technical Specifications	T.S. Basis	3.10	B.3.10-8	149

Facility Learning Objective:	P8188L-016 #6
Question Source:	New
Comments:	Although RO's are not typically required to know the basis of T.S. at PI, they are required to know why they are maintaining rods above the RIL which happens to be in the bases of the Prairie Island T.S.'s A conscious decision was made to use a negatively stated stem for this question. NUREG 1021 Appendix B sect C.2.e was referenced regarding the use of a negatively stated stem.

Record #	22
RO #	15
SRO #	

Stem:

Given the following conditions on unit 2:

- 21 and 24 Containment Fan Coil Units are running in fast to the gap
- 22 Containment Fan Coil Unit is running in slow to the dome
- 23 Containment Fan Coil Unit is tagged out-of-service
- A loss of offsite power then occurs causing the safeguards busses to transfer to the diesel generators
- When power is restored the ONLY Containment Fan Coil Unit to restart is 21

Based on this information, which 480-volt bus failed to regain power?

Answers/Distracters:

a. Bus 211

b. Bus 212

c. Bus 221

d. Bus 222

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-F	2	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-See (d) below. This answer could be selected if the operator doesn't know which busses belong to which train and/or which CFCUs belong to which train.
b	Incorrect-See (d) below. This answer could be considered if the operator believes that bus 212 is a "B" train power supply
c	Incorrect-See (d) below. This answer could be considered if the operator believes that the "B train" 22 and 24 CFCUs are powered from bus 221 which is a "B" train power supply.
d	Correct-21 and 23 CFCUs are powered from 480-volt bus 212. 22 and 24 CFCUs are powered from 480-volt bus 222. After power is restored to the safeguard busses and loads sequenced onto the diesels all CFCUs would be expected to restart except 23 which is out-of-service. Since 21 CFCU is the only to restart, it can be determined that bus 222 did not re-energize.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
013 Engineered Safety Features Actuation System (ESFAS)	K2.01	3.6	3.8

K/A Statement:	- Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguards Ventilation: Containment air Handling	P8180L-009H	II.A.4.d.1	16	1
Containment systems	B19	3.13.A	17-19	4

Facility Learning Objective:	P8180L-009H #4.a
Question Source:	Facility Exam Bank
Comments:	Question P8186L-008 010 minor wording changes to reflect plant operating requirements.

Record #	23
RO #	16
SRO #	

Stem:

Given the following conditions for Unit 1:

- Reactor power is 100%
- Bus 15 is being supplied by D1 Diesel Generator
- 11 CC Pump is running

If a SI signal occurs, what would be the status of the CC Pumps?

Answers/Distracters:

a. 11 CC Pump trips THEN BOTH 11 and 12 CC Pumps start when the load restoration Permissives are met.

b. 11 CC Pump trips THEN 11 CC Pump restarts when the load restoration permissive is met. 12 CC Pump starts ONLY if CC pump discharge pressure remains below 65 psig.

c. 11 CC Pump continues to run. 12 CC Pump starts when the load restoration permissive is met.

d. 11 CC Pump continues to run. 12 CC Pump starts immediately upon SI actuation.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-RI	2	2	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - Following an SI signal the CCW Pump breakers are tripped on load shed and then restarted on the load reject sequence when permissive generated. This occurs regardless of the source of power to the buses. With the bus supplied from the DG only, the DG breaker remains shut and load reject sequencing occurs.
b	Incorrect - 12 CC Pump starts whenever the load sequence permissive is generated, regardless of CCW discharge pressure.
c	Incorrect - (Both) Pumps trip on Load Reject sequence for SI.
d	Incorrect - (Both) Pumps trip on Load Reject sequence for SI and Pump 12 starts on the load reject sequence when permissive generated (not immediately upon the SI signal).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
003 Reactor Coolant Pump System	K2.02	2.5	2.6

K/A Statement:	Knowledge of bus power supplies to the following: - CCW pumps

Reference Title:	Reference #:	Section:	Page:	Rev:
Component Cooling	P8172L-002	V.C.7.a	32	4
4.16 KV STATION AUXILIARY SYSTEM	B20.5	3.6.4	10-11	3
Interlock Logic Diagram - Unit 1 Bus 15 Load Rejection-Restoration	NF-159024			C

Facility Learning Objective:	P8172L-002 #3, 5
Question Source:	facility initial exam bank question P8172L-002 026
Comments:	Changed premise to having one bus supplied from DG. Changed all selections to reflect CCW Pump 11 supplied from DG at SI signal generation and potential effects. Cognitive ranking is based on the fact that the candidate will have to recognize the interactions between the CC system, Safeguards 4160V system, and the SI actuation system and the consequences of these interactions.

Record #	**24**
RO #	**17**
SRO #	**18**

Stem:

Given the following conditions on Unit 1:

- Reactor power is 100%
- 11 RCP seal leakoff indication is offscale high
- The following annunciators are lit
  - **47012-0301**, 11 RCP STANDPIPE HI LVL
  - **47015-0206**, 11 RCP LABYRINTH SEAL LO DP
  - **47015-0306**, 11 RCP SEAL LEAKOFF HI FLOW

Which of the following has occurred AND what action is required?

Answers/Distracters:

a. 11 RCP #1 seal has failed AND an immediate (within 5 min) reactor trip is required.

b. 11 RCP #2 AND #3 seals have failed AND an immediate (within 5 min) reactor trip is required.

c. 11 RCP #2 seal has failed AND a controlled shutdown (within 8 hr) is required.

d. 11 RCP #1 AND #2 seals have failed AND a controlled shutdown (within 8 hr) is required.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a	2-RI	2	4	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - Low labyrinth seal DP, #1 seal flow offscale high is >8 gpm, and standpipe high level indicates #2 seal is starting to film ride as expected for a #1 seal failure. The # 2 seal is beginning to respond as a film riding seal.
b	Incorrect - A damaged #2 & #3 seal does not change the Labyrinth seal DP and will result in low standpipe level. Also Containment sump level and or RCDT level would be rising.
c	Incorrect - With the #2 seal failed seal leakoff flow has decreased indication and RCDT level would increase.
d	Incorrect - With the #2 seal failed seal leakoff flow has decreased indication and RCDT level would increase.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
003 Reactor Coolant Pump System	K6.02	2.7	3.1

K/A Statement:	Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: - RCP seals and seal water supply

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Reactor Coolant Pumps	P8170L-002	IV.A.7.h.1	15-16	2
Failure Of A Reactor Coolant Pump Seal	1C3 AOP3	2.4.4 Table	5	8

Facility Learning Objective:	P8170L-002 #6, 8
Question Source:	facility initial exam bank question P8170L-002 004
Comments:	Added alarm indications for flow & DP.

Record #	**25**
RO #	**18**
SRO #	**19**

Stem:

Given the following conditions on Unit 1:

- The Unit is at 80% power
- RCS boron concentration is 450 ppm
- CVCS Makeup control is in AUTO and set for makeup at 440 ppm
- All other control systems are in AUTO
- Rods begin to step in
- As rod motion stops, RCS Tavg is noted to continue to rise

What was the event that initiated this transient? **(Assume no operator action was taken).**

Answers/Distracters:

a. Power Range Channel N41 FAILED HIGH.

b. VCT level transmitter LT-141 FAILED LOW.

c. Controlling Pressurizer level channel LT-428A FAILED LOW.

d. Main turbine impulse pressure controller PM-485A FAILED HIGH.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
c	3-PEO	2	4	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect- failure of a PR range high will cause rods to begin to step in due to rate of change of power compared to Tref. Rod motion would quickly stop when the rate of change of power falls to zero. However Tavg would not be affected and would not continue to rise.
b	Incorrect- This may be expected if the level channel feeding makeup control fails low, as auto Makeup would be initiated and the dilution would occur at a slower rate (due to lower charging flow). However the channel listed does not feed the Makeup control.
c	Correct- A failed controlling Przr level channel will cause the Charging Pump in AUTO to go to high speed increasing charging flow to the RCS. Since charging flow exceeds letdown flow (which also isolates), The VCT level will drop. When auto makeup initiates at VCT level of 18%, sensed on LT-112, the makeup is added at a lower boron concentration. The reduction in boron concentration added to the RCS results in increasing Tavg. When Tavg exceeds Tref by 1.5°F the CRDS will generate a signal to drive control rods in to restore Tavg to Tref. At a difference of 1°F, rod motion will stop, but RCS Tavg continues to rise due to the dilution.
d	Incorrect- Failure of impulse pressure high results in Tref signal failure high. Since this is a positive input to rod control the systems responds by attempting to raise Tavg to match Tref. Rods would step out at maximum speed without stopping (until rod stop reached) not in. Also Tavg would have stabilized following stopping of rod motion (over time).

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
004 Chemical and Volume Control System	K3.01	2.5	2.9

K/A Statement:	Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: - CRDS (automatic)

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Pressurizer Level Control System	P8170L-006	III.C.3 V.C.1.a.2	10 20	3
CVCS	P8172L-001a	II.C.1.a.2	12-13	3
Interlock Logic Diagram Chemical Volume and Control System Unit 1	NF-40784-4			J

Facility Learning Objective:	P8170L-006 #1.a, 7.b P8172L-001a #3
Question Source:	NRC Exam Bank
Comments:	DC Cook 1998 SRO Exam question (page 49). Minor changes to adapt to facility controls and equipment.

Record #	26
RO #	19
SRO #	20

Stem:

Given the following conditions on Unit 1:

- Reactor power is 100%
- **CV-31202**, LTDN TEMP CONT, **fails closed** due to a controller malfunction.
- After approximately 2 minutes the local operator was able to correct the problem and normal flow was restored through **CV-31202.**
- LTDN temperature rises to 160°F AND then returns to normal.

Assuming NO other operator action was taken, how is normal letdown flow restored?

Answers/Distracters:

a. Manually **open CV-31203** using **1HC-135A,** LTDN PRESS CONT, UNTIL letdown pressure is approximately 275 psig, AND THEN **return 1HC-135A** to AUTO.

b. **Position** ONE Control Switch for **CV-31325,** OR **CV-31326** OR **CV-31327,** LTND ORIFICE ISOL, to OPEN, AND THEN **return** Control Switch to AUTO.

c. **Position** Control Switch for **CV-31204** "LTDN DIVERT TO PURIF" valve to DIVERT, AND THEN **return** Control Switch for **CV-31204** to AUTO.

d. **Position** Control Switch for **CV-31205,** LTDN DIVERT TO HOLDUP TNK, to the VC TANK position AND THEN **return** Control Switch for **CV-31204** to AUTO.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-RI	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect - Letdown pressure controller operates based on downstream pressure. With the CCW valve failed, pressure is not expected to change significantly and the valve will operate within normal parameters for the given time. No action is expected for this valve.
b	Incorrect – The orifice isolation valves are not affected by the given failure. These valves will auto close on low-low Pressurizer level. Pressurizer level is unaffected by the failure.
c	Correct - Applying full pressure to the CCW from Letdown HX vale will result in closure of the valve. This will stop CCW flow through the HX and result in rising letdown temperature. At 140°F, a high letdown flow alarm will actuate (47015-0408) and CV-31204 will go to the VCT position, bypassing the letdown demineralizers. When normal temperature is restored, flow through the demineralizers is reestablished by taking the control switch for CV-31204 to the DIVERT position and returning to AUTO.
d	Incorrect - The letdown divert valve operates on VCT level and diverts flow to the HUT(s) on high VCT level. VCT level is not affected by the failure.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
004	Chemical and Volume Control System	K4.03	2.8	2.9

K/A Statement:	Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: - Protection of ion exchangers (high letdown temperature will isolate ion exchangers)

Reference Title:	Reference #:	Section:	Page:	Rev:
CVCS	P8172L-001a	IV.4.f	23-24	3
COMPONENT COOLING SYSTEM	B14	3.8.B	13	4
Interlock Logic Diagram Chemical Volume & Control System U-1	NF-40784-1			K

Facility Learning Objective:	P8172L-001a #3
Question Source:	New question
Comments:	

Record #	27
RO #	20
SRO #	

Stem:

Given the following conditions on Unit 1:

- Cooldown is in progress for refueling
- RCS hot legs are at 180°F
- RCS pressure is 290 psig
- 12 RHR pump is in service
- Both RCPs are secured
- **CV-31236**, 12 RHR HX RC OUTLET FLOW, FAILS OPEN, resulting in an overcurrent trip of 12 RHR pump

Which of the following actions should be taken to restore core cooling?

Answers/Distracters:

a. **Establish** RCP support conditions AND **start** a RCP.

b. **Close** and partially open **MV-32066**, RHR TO RC LOOP B COLD LEG, **start** 11 RHR pump AND locally **throttle MV-32066**.

c. **Close RHR-2-5**, RHR HX Outlet Crossover valve, AND **start** 11 RHR Pump.

d. **Fully close CV-31237** (HC-626A), 11/12 RHR HX Bypass Flow valve, THEN **reset** AND **restart** 12 RHR Pump.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	3-SPK	2	3	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-RCP operation is not warranted at this time since Train A RHR is available and the RCS temperature is below 212°F
b	Correct- With CV-31236 failed full flow will be directed through RHR Hx 12, resulting in excessive RHR cooldown. Throttling the flowpath to the RCS, By closing the RHR return motor valve and throttling using the valve handwheel will allow flow from 11 RHR Pump to be controlled until CV-31236 can be repaired.
c	Incorrect- Closure of the RHR Hx Outlet Crossover valve(s) will only result in flow through the 12 RHR Hx. The 11 RHR pump will also trip on overcurrent. (NOTE: If both one of the Inlet Crossover and one of the Outlet Crossover valves were closed, Then 11 RHR Pump will NOT be able to discharge to the Loop B cold leg return and the RHR Pump will be deadheaded).
d	Incorrect- This will reduce overall system flow, however the cooldown rate cannot be controlled (all flow through 12 RHR HX). Also the 12 RHR Pump should not be restarted until it has been checked.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
005 Residual Heat Removal System (RHRS)	A4.01	3.6	3.4

K/A Statement: Ability to manually operate and/or monitor in the control room:
-Controls and indication for RHR pumps

Reference Title:	Reference #:	Section:	Page:	Rev:
Residual Heat Removal	P8180L-003	V.A.2, 3	37-38	3
Flow Diagram	X-HIAW-1-31			M
RHR Operation Without Control Room Instrumentation or Flow Control	1C15 AOP3	2.4.2	5	5

Facility Learning Objective:	P8180L-003 #2,9
Question Source:	Facility Exam Bank
Comments:	Initial bank question P8180L-003 011. Changed correct answer to reflect correct actions detailed in 1C15 AOP3.and minor wording changes.

Record #	28
RO #	21
SRO #	21

Stem:

Given the following conditions on Unit 1:

- A plant heatup is in progress following a refueling outage
- RCS temperature is 230°F
- Both RCPs are running
- 11 RHR pump is in service with RCS heatup being controlled using 11 RHR heat exchanger
- 11 CC surge tank level is +8 inches and rising
- 1R-39, CC SYSTEM LIQUID MONITOR, is indicating normally

Which of the following actions should be taken to correct this condition?

Answers/Distracters:

a. **Verify MV-32088,** 11 CC SURGE TANK VENT, is CLOSED.

b. **Close CV-31245** AND **CV-31246**, RCP THERM BARRIER CLNT OUTL.

c. **Verify** both CC pumps are operating AND **initiate** CC flow through 12 RHR heat exchanger.

d. **Open 1HC-624,** 11 RHR HX RC OUTLET FCV, to limit heatup rate.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c.	2-RI	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect – This would be a correct action if 1R-39 were in alarm per ARP
b	Incorrect – This would be correct if there was indication of a thermal barrier HX leak per C14 AOP2 and is again contradicted by the normal 1R-39 reading
c	Correct – RCS temperature is greater than 225 deg and therefore boiling is occurring in the off-service RHR HX causing an increase in CC surge tank level. 1C15 and 1C1.2 initiate this action
d	Incorrect – Although this action may decrease the severity of the boiling, it will not eliminate it unless RCS temp is decreased to less than 225 deg. It is also not directed by procedure

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
006 Emergency Core Cooling System (ECCS)	A1.12	2.9	3.4

K/A Statement:	Ability to predict and/or monitor changes in parameters associated with operating the ECCS controls including: -RHR heatup limits

Reference Title:	Reference #:	Section:	Page:	Rev:
Residual Heat Removal	P8180L-003	V.C	39-40	3
Residual Heat Removal System	1C15	3.6	4	22
Unit 1 Startup Procedure	1C1.2	5.4.8	40	24

Facility Learning Objective:	P8180L-003 #8,9
Question Source:	New question
Comments:	

Record #	29
RO #	22
SRO #	22

Stem:

Given the following conditions on Unit 1:

- A design-basis LOCA has occurred
- Train A SI actuation fails
- Train B SI actuation occurs as expected
- After 10 minutes, all Train A ECCS equipment is manually started

What is the result of these conditions?

Answers/Distracters:

a. Adverse Containment conditions will occur SOONER.

b. Train A ECCS equipment will NOT be available for recirculation operation.

c. Containment pressure will peak HIGHER than when both trains auto actuate.

d. Following depressing the SI RESET pushbuttons, SI CANNOT be manually reinitiated on EITHER train.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c.	3-PEO	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Adverse containment conditions will exist when containment pressure rises to and above 5psig or containment radiation rises to and above 1E4 R/hr. Containment pressure is expected to rise above 5 psig within the first couple seconds following the event. ECCS flow to the core will not occur until 25 to 28 seconds following the event (due to RCS blowdown).
b	Incorrect- Manually starting ECCS equipment has no affect on the future operation of the systems. Once started the ECCS equipment (RHR & SI Pumps) can be manually aligned (as normal) to Sump B.
c	Correct- Operation of one train of ECCS is assumed in the accident analysis and defines the limiting conditions for accident conditions. Core heat removal with both trains automatically operating will provide more fluid for heat removal quenching the core more quickly and limiting the rise of containment pressure.
d	Incorrect-Assuming the SI signal is not generated, the Safeguards Actuation Logic for the unaffected Train is not affected by the failed Train. B Train SI may always be manually actuated. The Block and Reset of automatic SI may be affected

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
006 Emergency Core Cooling System (ECCS)	K3.03	4.2	4.4

K/A Statement:	Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: -Containment

Reference Title:	Reference #:	Section:	Page:	Rev:
Emergency Core Cooling System	P8180L-005	VIII.C.2	33-35	2
E-1/E-2 Review	P8197L-012	VIII.B	50-51	2

Facility Learning Objective:	P8180L-005 #5,6 P8197L-012 #26
Question Source:	New
Comments:	

Record #	**30**
RO #	**23**
SRO #	**23**

Stem:

If CC flow to a RCP is lost, the RCP is tripped to prevent damage to which of the following components?

Answers/Distracters:

a. RCP motor bearing

b. RCP radial bearing

c. Motor stator windings

d. Thermal barrier heat exchanger

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a.	1-F	2	2	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Correct- CC is supplied for cooling of the lower guide bearing and thrust bearing/upper guide bearing for the RCP motor. Loss of CC leaves no cooling for these components.
b	Incorrect- The RCP radial bearing is cooled by seal injection normally or RCS if seal injection is lost (from Thermal Barrier HX).
c	Incorrect- The motor windings are air-cooled.
d	Incorrect- The Thermal Barrier HX is supplied with CC water for cooling. However, it limits heat transfer from RCS to pump seals and radial bearing with normal seal injection flows down past the heat exchanger. The HX cools the RCS flow up through heat exchanger in the event seal injection flow is lost.

K/A System/Evolution:		*K/A #:*	*KAVRO*	*KAVSRO*
008	Component Cooling Water System (CCWS)	K3.03	4.1	4.2

K/A Statement:	Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: -RCP

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Component Cooling	P8172L-002	IV.A.4.g.1	16	4
Reactor Coolant Pumps	P8170L-002	IV.B.3 & 4	18	2
LOSS OF COMPONENT COOLING	1C14 AOP1	2.3	4	8

Facility Learning Objective:	P8172L-002 #6.a P8170L-002 #8
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8170L-002 010.

Record #	31
RO #	24
SRO #	24

Stem:

Given the following conditions on Unit 1:

- RCS Tavg is 150°F
- RCS pressure is 280 psig
- Testing is in progress on Pressurizer pressure channel 1PT-449A with the channel bistables tripped
- OPPS is in ENABLE

If 1PT-419 wide range loop pressure transmitter FAILS HIGH, how do the pressurizer PORVs respond and what is the reason for this response?

Answers/Distracters:

a. Neither PORV opens because the coincidence is 2/2.

b. Only 1PCV-430 (PORV "A") opens because the failed transmitter is in its train.

c. Only 1PCV-431C (PORV "B") opens because the bistable to the PORV interlock is tripped.

d. Both PORVs open because the coincidence is 1/2.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	2-DR	2	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The interlock is 1/2.
b	Incorrect- The wide range pressure output for OPPS is not separated since each pressure transmitter inputs a signal to each PORV for operation.
c	Incorrect- Interlocks exist that prevent the PORVs from opening when the OPPS disabled (normal Pressurizer pressure control) in the event a PRZR pressure channel fails high. Also the coincidence for PORV operation is 1/2. With the interlock bistable tripped (1PC-449B) and the Pressurizer pressure control in AUTO, failure high of the controlling Pressurizer pressure channel (normally 3 - PT-431) would result in opening of PORV B.
d	Correct- With OPPS enabled and the PORV in AUTO, the PORV actuation coincidence is 1/2. PCV-430 and PCV-431C will open at 500 psig from PT-419 or PT-420.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
010 Pressurizer Pressure Control System (PZR PCS)	K6.01	2.7	3.1

K/A Statement:	Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: -Pressure detection systems

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Process Instrumentation System	P8184L-003	XII.A.2.d, 3.b.1	39-40	3
REACTOR COOLANT SYSTEM	B4A	3.7.1	20	5
Interlock Logic Diagram Pressurizer Pressure System Unit 1	NF-40780-1			L

Facility Learning Objective:	P8184L-003 #14
Question Source:	Facility Exam Bank
Comments:	Initial Bank Question P8170L-005 007. Modified 3 selections. Two choice (including answer changed to direct coincidence). Added condition for testing on Przr pressure channel and selection including interlock input for PORV opening (normal operations)

Record #	32
RO #	25
SRO #	25

Stem:

Given the following conditions on Unit 1:

- Reactor power is stable at 30%
- RCS Tavg is 551°F
- Pressurizer pressure is 2230 psig
- Pressurizer level is 25%
- Pressurizer Level Control Transfer Switch is in the NORMAL 2-3 position
- Charging Pump Speed Controller (**1LC-428F**) output FAILS LOW

What automatic action(s) will occur over time as a result of this failure assuming NO operator action is taken?

Answers/Distracters:

a. Pressurizer level will rise to 50% AND stabilize.

b. Pressurizer level will drop to 21% AND stabilize.

c. Letdown will isolate AND the reactor will trip on high Pressurizer level.

d. Backup heaters will energize AND the reactor will trip on high Pressurizer pressure.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c.	3-PEO	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- This is the maximum Pressurizer control level corresponding to failed temperature input (Tref) of 580°F.
b	Incorrect- This is the normal zero-power Pressurizer level setpoint.
c	Correct- When the output of the Charging Pump Speed Controller ails it is as if the controlling level channel high fails high in that charging flow falls to minimum. At 14.8% level, letdown isolates charging continues at minimum (52 gpm) and Przr level rises to high level trip setpoint.
d	Incorrect- This may occur if the output fails high. The Pressurizer level would rise and at 10% above program level, the backup heaters (Group D and Group E) would be energized. However the pressure control system would be able to control the pressure by opening spray valves and/or PORVs as necessary. Also the pressure rise would be somewhat limited by the insurge of the cooler RCS water as level rises.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
011 Pressurizer Level Control System (PZR LCS)	K1.04	3.8	3.9

K/A Statement:	
	Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following: -RPS

Reference Title:	Reference #:	Section:	Page:	Rev:
Pressurizer Level Control System	P8170L-006	IV.A.3.	13-14	3
REACTOR CONTROL SYSTEM	B7	3.4.2	22-24	4
Interlock Logic Diagram Pressurizer System Unit 1	NF-40780-2			H

Facility Learning Objective:	P8170L-006 #6,7
Question Source:	New
Comments:	

Record #	33
RO #	26
SRO #	26

Stem:

Given the following conditions on Unit 1:

- Reactor power is 100%
- Reactor trip breaker testing is being performed with Reactor Trip Bypass Breaker B (BYB) racked in AND CLOSED
- Both Reactor Trip Breakers (RTA and RTB) are CLOSED
- An electrician inadvertently OPENS DC panel breaker 16-2 "B Train DC To Reactor Switchgear Cabinet".

What is the expected reactor response to the above conditions?

Answers/Distracters:

a. The reactor will automatically trip. The Control Room Operators will enter E-0, "Reactor Trip or Safety Injection", to establish stable plant conditions.

b. The reactor must be manually tripped from the Control Room. The Control Room Operators will enter E-0 to establish stable plant conditions.

c. The reactor must be manually tripped locally by opening Reactor Trip Bypass Breaker B (BYB). The Control Room Operators will enter FR-S.1, "Response To Nuclear Power Generation/ATWS", to establish stable plant conditions.

d. The reactor will trip automatically if a train "A" trip signal comes in BUT may remain in operation for up to 4 hours with BYB closed provided stable plant conditions exist.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	2-RI	2	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This would be the case if A Train DC power was lost (or if BYA had been closed). Both RTA and BYB would open on UV coils
b & c	Incorrect- B) & C) - Both choices use the difference between the reactor trip breakers and bypass breakers to lead to the conclusion that one of the breakers does not function normally. The difference referred to is the use of an additional set of trip relays for the reactor trip breakers - the shunt coils. In the case given this erroneously assumes the UV coils do not function and the shunt coils do work, opening the trip breaker only. For B) - It is assumed that the manual trip is required to open RTA in order to trip the reactor. For C) it is assumed the loss of DC allows no operation and local action must be taken to open the bypass breaker.
d	Correct- Bypass breaker B is operated from Train A logic and powered from Train A DC/AC. Trip Breaker B will open due to the operation of the UV trip coils. Trip Breaker A remains closed and the CRDMS remains energized.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
012 Reactor Protection System	A2.07	3.2	3.7

K/A Statement:	
	Ability to (a) predict the impacts of the following on the Reactor Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: -Loss of dc control power

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Protection	P8184L-004	C.3, 4	40-43	5
Logic Diagram Reactor Trip Signals Unit 1 & 2	X-HIAW-1-236			D

Facility Learning Objective:	P8184L-004 #8c
Question Source:	New
Comments:	

Record #	34
RO #	27
SRO #	27

Stem:

Given the following conditions on Unit 2:

- Reactor power is 30%
- Pressurizer pressure Yellow Channel (PT-449) has FAILED LOW AND was removed from service in accordance with 2C51 "Instrument Failure Guide - Unit 2"

Which of the following additional bistable actuations in this condition would result in a reactor trip?

Answers/Distracters:

a. White Channel Overtemperature Delta-T **2TC-405-C,** OVER TEMP TRIP

b. Blue Channel Turbine Impulse Pressure **2PC-486-A,** TURBINE PRESS P13

c. Blue Channel Overpower Delta-T **2TC-407-A,** OVER POWER TRIP

d. Yellow Channel Nuclear Power Range Instrument Drawer **N44A,** OVERPOWER TRIP HIGH RANGE

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a.	3-PEO	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct- For the PT-449 failure the following bistables were tripped: 1) 408C OVER TEMP TRIP; 2) 408D ROD STOP; 3) 449A LO PRESS TRIP. Failure of 405C OVERTEMP TRIP bistable will result in 2/4 coincidence (405C & 408C) being satisfied for the OTdT trip.
b	Incorrect-486A feeds the P-13 permissive that inputs to the P-7 permissive for the at-power trips blocking. This permissive is usually active at turbine load > 10%. The failure therefore does NOT affect operations until power is decreased below this value, when the at-power trips would NOT be blocked
c	Incorrect- This does NOT result in meeting any coincidence for reactor trip. OPdT is similar to OTdT except that the pressure and dI inputs are zero (NOT used) in calculating the value.
d	Incorrect- This is in the same channel as the failed instrument so bistable actuation should not result in meeting the required coincidence. Note that while the affected bistable is NOT associated with any tripped for the failed pressure channel, failure of N44 itself would require tripping the OTdT bistables (408C & 408D).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
012	Reactor Protection System	A3.02	3.6	3.6

K/A Statement:	Ability to monitor automatic operations of the Reactor Protection System including: -Bistables

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Protection	P8184L-004	VII.A.4.b, VII.B.2	33, 35-36	5
Instrument Failure Guide - Unit 2	C51	1P-449-low	1-2	14
Logic Diagrams Primary Coolant System	X-HIAW-1-239			B

Facility Learning Objective:	P8184L-004 #6,8
Question Source:	New
Comments:	

Record #	35
RO #	28
SRO #	

Stem:

Given the following conditions on Unit 1:

- Excess letdown has been placed in service
- Normal letdown has NOT been isolated yet
- A spurious Containment Isolation actuation signal has occurred

Which of the following is the effect of the Containment Isolation signal on the CVCS system?

Answers/Distracters:

a. RCP seal injection flow is lost.

b. Letdown flow is diverted to the CVCS HUT.

c. Excess letdown flow is diverted to the RCDT.

d. RCP No. 1 seal leakoff flow is directed to the PRT via a relief valve.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
d	2-RI	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect - RCP seal injection flow is not affected by CI signal.
b	Incorrect - Letdown flow is isolated when the Letdown Line Containment Isolation Valve closes due to CI signal. CV-31205, Letdown Diversion form VCT to HoldUp Tanks valve, is affected by high VCT level.
c	Incorrect - Excess letdown and #1 seal return go to the PRT. CV-31334, Excess Letdown to Drain Tank, continues to be aligned to VCT. On a loss of air to the valve operator results in failure to VCT.
d	Correct - When the CI signal occurs, among other actions, The Excess Letdown (and Seal Water Return) Isolation Valves go close. This results in pressurization of the line inside containment and opening of Relief Valve VC-25-1 which directs flow to the PRT.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
013 Engineered Safety Features Actuation System (ESFAS)	A4.03	4.5	4.7

K/A Statement:	Ability to manually operate and/or monitor in the control room: - ESFAS initiation

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Engineered Safeguards System	P8180L-006	V.B.2	19	3
CVCS	P8172L-001a	IV.A.2.b	14-15	3
Interlock Logic Diagram - Chemical Volume & Control System - Unit 1	NF-40784-2			H

Facility Learning Objective:	P8180L-006 #3; P8172L-001a #4,9
Question Source:	New
Comments:	

Record #	36
RO #	
SRO #	28

Stem:

(see reference)

Given the following conditions on Unit 1:

- Reactor power is 80%
- Control Bank 'D' demand position is 185 steps
- Following a control rod exercise at 0200, rod position indication (RPI) for a Control Bank C rod was 200 steps
- Actual rod position was determined to be 225 steps
- At 0400 the following rod position indication were noted for Control Bank D rods after demand position was changed to 200 steps:
    - G-3 RPI, 200 steps
    - C-7 RPI, 185 steps
    - G-11 RPI, 200 steps
    - K-7 RPI, 175 steps
- It is determined all Control Bank D rods are at 200 steps

What ACTION would be required?

Answers/Distracters:

a. Immediately **trip** the reactor AND enter E-0, Reactor Trip and Safety Injection.

b. **Reduce** reactor power to less than 50% by 1200 that day AND maintain power below 50% until all RPIs are repaired.

c. **Place** the Unit in HOT SHUTDOWN by 1000 that day if EITHER ALL Bank D RPIs OR the Bank C RPI is NOT repaired.

d. **Place** the Unit in HOT SHUTDOWN by 1000 the next day if ONE Bank D RPI is NOT repaired.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPR	2	4	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This action would be required only if a rod is dropped (RPI < 20 steps).
b	Incorrect-- If one rod position indication per group (for one or more groups) occurs, one ACTION is to reduce power to < 50% within 8 hours. If this were a correct action it would have to be taken 8 hours following the "C" Bank rod RPI failure (at 1000). An alternative ACTION is to verify the position of the rod indirectly using MIDS at least once per 8 hours. Note that this occurred to determine actual rod position.
c	Incorrect- This assumes action is required for the failure in two groups (more than one group) versus action required for failure of more than one RPI in a group. The timing is not consistent with possible actions but occurs without any repair time (24 hours).
d	Correct- Rod Position Indication is inoperable if: 1) Below 30 steps and above 215 steps demand position, the difference between the group demand position and RPI is > 24 steps. 2) Between 30 steps and 215 steps demand position, the difference between the group demand position and RPI is > 12 steps. All three RPIs are then inoperable. With more than one RPI per group inoperable (for one or more groups), restore the inoperable RPIs to OPERABLE within 24 hours such that a maximum of one rod position indicator per group is inoperable.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
014	Rod Position Indication System (RPIS)	2.4.48	3.5	3.8

K/A Statement:	Emergency Procedures / Plan -Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

Reference Title:	Reference #:	Section:	Page:	Rev:
Rod Control & Rod Position Indication	P8184L-005	IX.F	56	2
Prairie Island Technical Specifications		3.10.F.3.e	TS.3.10-6A	139

Facility Learning Objective:	P8184L-005 #9
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Initial Bank question P8184L-005 071. Changed question for rod misposition question to multiple RPI failure with TS application. Change 3 selections including correct answer Provide T.S. 3.10.F only as reference.

Record #	37
RO #	
SRO #	29

Stem:

With Unit 1 in REFUELING Mode, which of the following conditions, if it occurs, requires entry into C1.6 AOP1 "CONTAINMENT EVACUATION"?

Answers/Distracters:

a. A VALID Source Range At Shutdown High Flux Level alarm.

b. A stuck fuel assembly in the transfer tube.

c. A loss of audible count rate circuit in the control room.

d. Loss of the high voltage power supply for one Source Range channel while placing a fuel assembly in the core.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a.	1-P	2	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- With verification of the SOURCE RANGE AT SHUTDOWN HI FLUX LVL condition, the operator is directed to evacuate containment per C1.6 AOP1.
b	Incorrect- A stuck fuel assembly in the transfer tube requires entry into D5.2 AOP2 and a subsequent evacuation of the containment annulus.
c	Incorrect-Loss of audible count rate in containment requires that core alterations cease until audible counts in containment are restored.
d	Incorrect- Loss of one SR monitor requires that core alterations cease (the fuel assembly should not be seated in the core) until two SR channels are restored to service (with audible indication in containment).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
015 Nuclear Instrumentation System	2.4.4	4.0	4.3

K/A Statement:	Emergency Procedures / Plan: -Ability to recognize abnormal indications for system operating parameters which are entry-level

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	IV.A.5.a.2	18	4
Alarm Response Procedure	C47013	47013-0301	1	26
CONTAINMENT EVACUATION	C1.6 AOP1	2.1.1	2	6

Facility Learning Objective:	P8184L-002 #8
Question Source:	New
Comments:	

Record #	38
RO #	29
SRO #	30

Stem:

What is the response expected for the Source Range NIS audio output during a LOCA as the reactor vessel downcomer voids?

Answers/Distracters:

a. The count rate will RISE.

b. The count rate will DROP.

c. The count rate will REMAIN THE SAME.

d. The count rate will Initially RISE then quickly DROP as steam fills the downcomer.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	3-PEO	2	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- As voiding begins to occur in the downcomer, the water of higher density is replaced by steam, and the neutron leakage increases. Since more neutrons reach the SR detectors (located next to the reactor vessel wall) the count rate will increase as indicated by the increased rate in the audio.
b	Incorrect-- This could be considered but the leakage rate increases so the number of events detected increase.
c & d	Incorrect- This could be considered, particularly if the person fails to consider that the downcomer acts as a shielding medium between the core (source of neutrons) and the detector, but the leakage rate increases so the number of events detected increase.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
015	Nuclear Instrumentation System	A3.05	2.6	2.7

K/A Statement:	Ability to monitor automatic operations of the Nuclear Instrumentation System including: -Recognition of audio output expected for a given plant condition

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	IV.A.1.d) Dev. Question 14	12 75	4

Facility Learning Objective:	P8184L-002 #3
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8184L-002 053. Modified indication to audible count rate. Replaces 015 K1.04 (Both). KA selected from same System KA on replacement outline.

Record #	39
RO #	30
SRO #	

Stem:

Given the following conditions on Unit 1:

- The plant is at 100% power
- Control Bank D rods are at 210 steps withdrawn
- An electrical failure DE-ENERGIZES Panel 113 Instrument Bus III (Blue)

If the operator attempts to withdraw rods, which of the following prevents rod motion?

Answers/Distracters:

a. Overpower Delta-T Rod Stop.

b. Overtemperature Delta-T Rod Stop.

c. Power Range Overpower Rod Stop.

d. Intermediate Range High Flux Rod Stop.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c.	1-F,I	2	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Overpower Delta-T may be affected since power range is an input to the OP calculation. However the input is delta-flux (top - bottom) which should be equal- resulting in ZERO input. If the Rod Stop bistable is affected, the coincidence is 2/4 to block rod withdrawal.
b	Incorrect-(As in A)) Overtemperature Delta-T may be affected since power range is an input to the OT calculation. However the input is delta-flux (top - bottom) which should be equal - resulting in ZERO input. If the Rod Stop bistable is affected, the coincidence is 2/4 to block rod withdrawal.
c	Correct- If 1 of 4 channels of power range overpower bistable actuates, a rod stop is generated. Panel 113 is the power for Channel III Instruments so Power Range N43 bistables trip.
d	Incorrect- The IR channels are NOT affected since power to them is either Channel I (N35) or Channel II (N36). (IR Rod Stop is normally blocked above P-10)

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
015 Nuclear Instrumentation System	K2.01	3.3	3.7

K/A Statement: Knowledge of electrical power supplies to the following:
-NIS channels, components, and interconnections

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	IVC.2.e.1.c, 3.a	28-30	4
Logic Diagram Nuclear Instr Permissives & Blocks	X-HIAW-1-238			B

Facility Learning Objective:	P8184L-002 #6,13
Question Source:	NRC Exam Bank
Comments:	Kewaunee 2000 NRC exam. Change power supply identification for plant difference

Record #	**40**
RO #	**31**
SRO #	**31**

Stem:

The following conditions exist on Unit 1:

- Reactor power is 100%.
- All control systems are in automatic
- Red Pressurizer Pressure Channel (PT-429) was declared inoperable and taken out of service with the appropriate bistables placed in the tripped condition.
- Controlling Pressurizer Pressure Channel (PT-431) fails HIGH.

What is the expected plant Response to the channel failure? **(assume NO operator action)**

Answers/Distracters:

a. BOTH PORVs AND BOTH spray valves OPEN resulting in a reactor trip from low Pressurizer pressure followed by SI actuation.

b. The reactor will TRIP on high pressure, AND SI will ACTUATE on low pressure due to spray valve operation.

c. Pressurizer proportional heaters will de-energize AND spray valves will OPEN resulting in an OTΔT runback prior to the reactor tripping, AND SI will ACTUATE due to low Pressurizer pressure.

d. BOTH PORVs AND BOTH spray valves remain CLOSED while Pressurizer heaters de-energize.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
b	3-PEO	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect – It requires a 2/2 interlock to open the PORVs in automatic. PORV PCV-430 takes a signal from red channel PT-429 and a signal from the interlock channel PT-430.. PORV PCV-431C requires a signal from yellow channel PT-449 and a signal from the controlling channel PT-431. With PT-449 and PT-430 not failed the PORVs do not make up the required logic to actuate and they remain closed. Also, although both spray valves open the reactor would trip directly on a high pressure signal.
b	Correct-With Red Channel bistables tripped and a second channel failing high results in 2 high pressurizer pressure bistables actuated which would cause a reactor trip. The spray valves will modulate fully open in response to the controlling pressure signal failing high. Subsequently actual pressure decreases until SI is actuated on low pressurizer pressure.
c	Incorrect-This answer could be considered if the candidate doesn't realize that the reactor will trip directly from 2/4 high-pressure signals.
d	Incorrect- Both PORVsdo remain closed and the heaters will de-energize but the sprays will open when the controlling channel fails high.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
016 (NNIS) Non-Nuclear Instrumentation System	K3.08	3.5	3.7

K/A Statement:	Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: -PZR PCS

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Pressurizer Pressure Control System	P8170L-005	III.B, V.C.1, VII.E	8-9, 25-26,39	4
Reactor Control System	B7	3.3	16-22	4
Reactor Control System	B7	Figure B7-14	54	4
Instrument Failure Guide	1C51.1	PT-429 high failure	1-2	14

Facility Learning Objective:	P8170L-005 #4,5
Question Source:	Facility exam bank question P8170-005 002
Comments:	

Record #	41
RO #	32
SRO #	32

Stem:

Given the following conditions on Unit 1:

- Reactor power is 100%
- Red channel Thot RTD detector developed an open circuit
- Operator action has stabilized the plant

What function is disabled when the Red Channel is selected on the Tavg Defeat Switch AND taken to Pull Out?

Answers/Distracters:

a. The continuous auto rod withdrawal signal.

b. The high Pressurizer level control signal.

c. The OPΔT rod stop channel alert.

d. The Tavg deviation alarm.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F	2	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-The failed high (control) signal to the rod control is removed. However the rods are demanded to INSERT with high Tavg since Tref from the turbine impulse pressure instrument provided the reference temperature.
b	Correct-The DEFEAT switch is used to remove the control signal input from the failed channel to the temperature-related control functions. One of the control functions is the auctioneered high temperature signal used for generating the Przr reference level signal. This control circuit is separate from the protection circuits.
c	Incorrect-- OPDT rod stop is a part of the protection circuit and so is separated from the control function. The DEFEAT switches are downstream from the ISO amps that provide separation of the protection and control circuits.
d	Incorrect- The inputs for each channel Tavg to the deviation alarm is upstream of the DEFEAT switches.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
016	Non-Nuclear Instrumentation System (NNIS)	K5.01	2.7	2.8

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to the NNIS: -Separation of control and protection circuits

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Process Instrumentation System	P8184L-003	IVA.1, E.1	14-15, 25	3
Logic Diagram Rod Control & Rod Blocks	X-HIAW-1-243			B

Facility Learning Objective:	P8184L-003 #5,6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8184L-003 006. Editorial change to question only

Record #	**42**
RO #	**33**
SRO #	**33**

Stem:

Given the following conditions on Unit 2:

- A loss of offsite power with reactor trip has occurred.
- 2EMB is DEENERGIZED due to a fault on the bus.
- Inadequate Core Cooling Monitor (ICCM) Train A was OOS prior to the trip.

How are core conditions indicating Natural Circulation flow verified in 2ES-0.1, "Reactor Trip Recovery", Attachment A?

Answers/Distracters:

a. The ERCS displays for Subcooling AND CETC's on Train A.

b. Subcooling from the Train A Subcooling monitor, CETC temperatures by local readings on the junction boxes.

c. Subcooling by comparing HIGHEST hot leg temperature to RCS wide range pressure, CETC temperatures from ERCS display Train B.

d. Subcooling by comparing HIGHEST hot leg temperature to RCS wide range pressure, CETC temperatures by local readings on the junction boxes.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a.	1-F,P	2	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- Both trains of ICCM are unavailable. "B" Train ICCM is powered from 2EMB. CETC and wide range pressure inputs to ERCS are independent (while RVLIS D/P transmitter information is output from ICCM).
b	Incorrect- Represent other possible sources of information that are not on Attachment A.
c	Incorrect- Represent other possible sources of information that are not on Attachment A.
d	Incorrect- Represent other possible sources of information that are not on Attachment A.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
017 In-Core Temperature Monitor (ITM) System	K3.01	3.5	3.7

K/A Statement:	Knowledge of the effect that a loss or malfunction of the ITM System will have on the following: -Natural circulation indications

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
ICCM	P8170L-001A	III.C, D, IV.A.11	10-11, 24	2
EMERGENCY RESPONSE COMPUTER SYSTEM	B41	3.5.3	10-13	2
INCORE INSTRUMENTATION SYSTEM	B10	4.2,	18-19	2

Facility Learning Objective:	P8170L-001A #2b, c
Question Source:	NRC Exam Bank
Comments:	Prairie Island 2000 NRC exam, question # 50. Editorial changes to question and one selection

Record #	43
RO #	34
SRO #	34

Stem:

Given the following conditions on Unit 1:

- A LOCA is in progress
- Average of core exit thermocouple readings is 434°F
- RCS wide range pressure readings are 275 psig
- Containment pressure is 8 psig

What is the status of core subcooling as displayed on the ICCM?

Answers/Distracters:

a. 16-18°F subcooled.

b. 19-20°F subcooled.

c. 16-18°F superheated.

d. 19-20°F superheated.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPR	2	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This may be considered if the sign associated with subcooling and superheat are transversed and the input of containment pressure is considered to impact this value (Adverse CNMT).
b	Incorrect- This may be considered if the sign associated with subcooling and superheat are transversed.
c	Incorrect- This may be the expected value if the input of containment pressure is considered to impact this value (Adverse CNMT).
d	Correct- The saturation temperature at 275 psig is 414°F. With an actual temperature at 434°F, the condition is superheat of 20°F. This is displayed on the ICCM as " 20°F superheated". Both the temperature detectors and the wide range pressure detectors are not affected by the containment conditions.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
017 In-Core Temperature Monitor (ITM) System	K5.03	3.7	4.1

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to the ITM System: -Indication of superheating

Reference Title:	Reference #:	Section:	Page:	Rev:
ICCM	P8170L-001A	IV.A.6; C.2,3,	15-16, 34	2
REACTOR COOLANT SYSTEM	B4A	3.4.3	15-16	5

Facility Learning Objective:	P8170L-001A #2.c, 6.b
Question Source:	Significantly modified from Facility Exam Bank values changed
Comments:	Incorporates the conditions from two questions P8170L-001A 001 and 014. Requires calculation of subcooling/superheat value and understanding that containment pressure does not affect inputs to Subcooling Monitor.

Record #	44
RO #	35
SRO #	

Stem:

Given the following initial conditions on Unit 1:

- Reactor power is 100%
- 12 CFCU is aligned to the dome
- 13 CFCU is aligned to the gap
- 14 CFCU is out of service

Then the following conditions are noted:

- Annunciator **47019-0406,** 13 CONTAINMENT FAN COIL UNIT MOTOR STATOR HI TEMP, actuates
- Damper **CD-34076,** 13 FCU DISCH TO DOME, indicates closed
- Damper **CD-34077,** 13 FCU DISCH TO GAP, indicates closed
- White light **ML-44002-0201**, 13 CNTMT FCU DISCH DMPRS IMPROPER, is lit

What action should be taken by the operator?

Answers/Distracters:

a.
- **Verify** 11 CFCU running in SLOW SPEED
- **Initiate** a Unit shutdown within EIGHT hours

b.
- **Take** 13 CFCU to STOP
- **Verify** 12 CFCU running in FAST SPEED
- **Open** damper **CD-34075**, 12 CFU DISCH TO GAP

c.
- **Verify** 13 CFCU running in FAST SPEED
- **Open** Damper **CD-34076,** 13 FCU DISCH TO DOME
- **Take** 13 CFCU control to SLOW SPEED

d.
- **Take** 13 CFCU control to SLOW SPEED
- **Open** Damper **CD-34077**, 13 FCU DISCH TO GAP
- **Return** 13 CFCU to FAST SPEED

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	3-SPK	2	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- per the Alarm response, the FCU is not considered inoperable even if the FCU is stopped if opening the damper does not result in stator temperature decrease. The controlling factor in this condition for TS is Containment temperature (shall not exceed 44°F above average air temperature in the shield building). If the FCU was considered inoperable then TS 3.3.B.2 applies and TS 3.0.3 with ONE-hour action applies.
b	Incorrect- Procedure limits operation of only one pair of fans to gap/support. Should never run 2 CFCUs to the same duct (I.e., 11 and 12 to the gap simultaneously) due to design pressure limitations of the duct work.
c	Incorrect-- Per procedure, Containment fan coil units that are used to supply air to the gap ductwork booster fans must be operated in FAST speed.
d	Correct- Per alarm response verifies one of the dampers open. Normal operation of the fans is in pairs with the 11 and 13 fans operating, the 12 and 14 fans operating as pairs. The 11 (or 12) will be running in FAST supplying the reactor vessel support cooling, and the 13 (or 14) fan running in FAST to the reactor gap cooling. With 14 fan OOS, the 12 fan would be running, (if required) supplying flow to the containment dome, in SLOW speed if possible (providing the desired containment temperature can be maintained.) Both dampers for the 13 CFU results in the alarm condition and the off-normal indication. The corrective action is to reopen one of the dampers. Since the 13 FCU only is available it would be required to supply the GAP. In order to align the damper, the fan must be running in LOW speed. The fans supplying the GAP are normally run in FAST speed. The other FCU is aligned to dome and is run in SLOW if possible (providing the desired containment temperature can be maintained.).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
022	Containment Cooling System	A2.03	2.6	3.0

K/A Statement: Ability to (a) predict the impacts of the following on the CCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
-Fan motor thermal overload/high-speed operation

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguards Ventilation: Containment Air Handling System	P8180L-009H	II.A.4.b)	11-15	1
Alarm Response Procedures	C47019	47019-0406	1	28
CONTAINMENT SYSTEM VENTILATION UNIT 1	1C19.2	3.7	4	15

Facility Learning Objective:	P8180L-009H #2.a, 8
Question Source:	New
Comments:	

Record #	**45**
RO #	**36**
SRO #	**35**

Stem:

Given the following conditions on Unit 1:

- Reactor power is 100%
- All CFCUs are in operation supplied by chilled water
- 11 and 21 Aux. Bldg. and Cont. Chillers and pumps are running split
- An inadvertent Train B (only) Safety Injection (SI) signal is generated

What is the expected status of cooling supplied to the CFCUs? **(Assume NO operator action is taken)**

Answers/Distracters:

a. 21 Chilled Water Pump is supplying 12 and 14 CFCUs, AND 121 and 11 Cooling Water Pumps are supplying 11 and 13 CFCUs.

b. 11 Chilled Water Pump is supplying 11 and 13 CFCUs, AND 21 and 22 Cooling Water Pumps are supplying 12 and 14 CFCUs.

c. 21 Chilled Water Pump is supplying 11 and 13 CFCUs, AND 121, 11 and 22 Cooling Water Pumps are supplying 12 and 14 CFCUs.

d. 11 Chilled Water Pump is supplying 12 and 14 CFCUs, AND 21 and 22 Cooling Water Pumps are supplying 11 and 13 CFCUs.

Answer:	***LOK***	***Tier:***	***LOD***	***RO Group:***	***SRO Group:***	***Facility:***	***Exam Date:***
b.	2-RI	2	4	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-. A) & C) - Use the Chilled Water system as the safety-related supply and assumes cross-ties activated in manner similar to Cooling Water header valves (Train B aligned to Unit 2 Pump. A) & D) - Also reverse the order of water supply to CFCUs
b	Correct- The Cooling Water System is normally supplied by motor driven pumps 11 & 21 with the headers cross-tied between Units. Upon SI, Chilled Water is isolated from the CFCUs and the Cooling Water valves aligned to supply the CFCU coils. Additionally, in the Cooling Water System, the Diesel Driven Pumps 12 and 22 , and the swing pump 121 auto start. A pair of header isolation valves close separating Units and aligning the 121 Pump to the Unit with the SI. A Train SI signal starts the 12 Pump while a B Train SI starts the 22 Pump. The 121 Pump will stop when both the 12 and the 22 Pumps reach rated speed. The Train B components are associated with 12 and 14 CFCUs. Thus in this situation the 121 Pump starts and runs. The 22 Pump starts and runs. The header is separated with the 121 Pump aligned to Unit 1 when MV-32037 (Train B valve) closes. Thus the Train A CFCUs are supplied from the normal Chilled Water System and the Train B CFCUs supplied from the Train B Cooling water Header with the 21 and 22 Cooling Water Pumps running.
c	Incorrect- C) & D) - Improperly have the 21 chilled Water Pump supplying the Unit 1 header.
d	Incorrect-See Basis for A and C above.

K/A System/Evolution:		***K/A #:***	***KAVRO***	***KAVSRO***
022	Containment Cooling System (CCS)	A4.02	3.2	3.1

K/A Statement: Ability to manually operate and/or monitor in the control room: -CCS pumps

Reference Title:	***Reference #:***	***Section:***	***Page:***	***Rev:***
Cooling Water System	P8176L-003	VIII.A	59-60	3
Safeguards Ventilation: Containment Air Handling System	P8180L-009H	II.A.2.d.5	26	1
Flow diagram Cooling Wtr Sys unit 1 AUX Bldg	NF-39216-3			R
Interlock and logic diagram Cont & AUX Bldg Chilled Water Sys	NF-86186-4			H
Interlock and logic diagram Cooling Water System - Units 1 & 2	NF-40315-1, 2, 11			S, N, B

Facility Learning Objective:	P8176L-003 #3,7 P8180L-009H #3
Question Source:	New
Comments:	

Record #	**46**
RO #	**37**
SRO #	**36**

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- Containment Spray has actuated
- RWST level currently reads 40%
- CA to CS **31941** and **31938** status lights are NOT lit on the SI active panel
- Caustic Addition Standpipe level currently reads 100%

What action must be taken by the operator to establish flow from the CA Standpipe to the Containment Spray System AND what is the effect if this action is NOT taken?

Answers/Distracters:

a. The operator must **open** EITHER **CV-31938** OR **CV-31941,** CAUSTIC ADDITION VALVE to establish flow. If the action is not taken the concentration of radioactive iodine in containment atmosphere will be higher.

b. The operator must **open** BOTH **CV-31938** AND **CV-31941**, CAUSTIC ADDITION VALVES, to establish flow. If the action is not taken containment pressure peaks at a higher value due to the reduced heat removal capacity of the CS spray.

c. The operator must **start** the CARF Pump AND **open** EITHER **CV-31938** OR **CV-31941**, Caustic Addition valve, to establish flow. If the action is not taken corrosion of components in containment increases due to higher pH value of the containment sump fluid.

d. The operator must **start** the CARF Pump AND **open** BOTH **CV-31938** AND **CV-31941**, Caustic Addition valves, to establish flow. If the action is not taken containment radiation levels are higher due to the increased radioactive noble gas production.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a.	2-RI	2	3	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- Consequences of interaction between Caustic Add and CS system post accident and action that must be taken to mitigate consequences. The addition of NaOH to the fluid increases its pH resulting in greater iodine absorption. A LOCA could result in the release of radioactive iodine gas, a fission product. The increased pH of the spray fluid causes elemental iodine to react and transform to a water soluble form. The iodine then goes into solution, removing it from the containment atmosphere. The increased pH of the fluid also minimizes the long term effects of corrosion on equipment in containment. The NaOH solution is gravity fed at a fixed ratio to the RWST flow at the suction of the pumps. Since there is a density difference between the RWST water and the NaOH [specific gravity (s.g.) of 1.0 to s.g. of 1.1], the level of water in the RWST must be maintained above the standpipe level to maintain the proper flow proportion. The header contains two parallel control valves, CV-31938 and CV-31941, which open automatically upon receipt of either an automatic 'P' signal or a manually initiated 'P' signal.
b	Incorrect- It is not required to open both valves since they are in parallel and the downstream header supplies both CS Pump suction piping. The failure to add NaOH does not affect the heat removal capability of the spray.
c	Incorrect- The CARF pump is normally only operated during filling the standpipe to ensure NaOH concentration by recircing the standpipe. While the corrosion of containment components is impacted by the addition of NaOH, the failure to add NaOH will result in a lower pH value due to the boric acid in the RWST.
d	Incorrect- The CARF pump is only operated during filling the standpipe to ensure NaOH concentration by recircing the standpipe. Noble gas concentration and its associated radioactivity level is unaffected by the addition of NaOH.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
026 Containment Spray System (CSS)	A2.05	3.7	4.1

K/A Statement:	Ability to (a) predict the impacts of the following on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: -Failure of chemical addition tanks to inject

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Containment Spray System	P8180L-002	III.A.2.a&f	14, 16-17	4
CONTAINMENT SPRAY SYSTEM	B18D	3.2.A	8-9	4
Flow Diagram Units 1 & 2 Caustic Addition System	NF-39252			P
Reactor Trip or Safety Injection	1E-0	step 5 RNO	5	19

Facility Learning Objective:	P8180L-002 #5,8
Question Source:	New
Comments:	

Record #	47
RO #	38
SRO #	37

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- The feeder breaker to 480V Bus 121 tripped open upon the LOCA
- Containment Pressure is 25 psig

What is the status of the Containment Spray system? **(Assume NO operator actions taken)**

Answers/Distracters:

a. 12 CS pump is running normally, 11 CS Pump is running at shutoff head.

b. 11 CS pump is running normally, 12 CS Pump is running at shutoff head.

c. 11 CS pump is running normally, 12 CS Pump is NOT running.

d. 12 CS pump is running normally, 11 CS Pump is NOT running.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F,S	2	3	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Correctly identifies both Pumps start but incorrectly identifies the discharge MV that didn't open.
b	Correct- Power supply to the CS Pump discharge valves are supplied from bus 111 through 1K1 (11 CS Pump disch) and bus 121 through 1KA2 (12 CS Pump disch). The valves are normally closed and open on the "P" signal. The valve operation is independent of the operation of the CS Pump, which should start on the "P" signal. The CS Pump discharge header is not automatically set up with mini-flow or recirculation capability.
c	Incorrect-Incorrectly identifies the 12 CS Pump as not running. The position of discharge valve does not affect operation of CS pump breaker.
d	Incorrect- Incorrectly identifies the 11 CS Pump as not running. The position of discharge valve does not affect operation of CS pump breaker.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
026	Containment Spray System (CSS)	K2.02	2	2

K/A Statement:	Knowledge of electrical power supplies to the following: -MOVs

Reference Title:	Reference #:	Section:	Page:	Rev:
Containment Spray System	P8180L-002	III.B.2	24-25	4
CONTAINMENT SPRAY SYSTEM	B18D	4.2	10-11	4
Interlock Logic Diagram	NF-40320-1			M

Facility Learning Objective:	P8180L-002 #4,9
Question Source:	New
Comments:	

Record #	48
RO #	39
SRO #	38

Stem:

Which of the following describes the response of the Containment Cleanup fans following a loss of offsite power?

Answers/Distracters:

a. The fans LOSE all sources of power on a loss of offsite power AND CANNOT be started.

b. The fans are PROVIDED with power by diesel generators AND can be manually started.

c. The fans are PROVIDED with power by diesel generators AND will automatically start if a SI signal is generated.

d. The fans are PROVIDED with power by diesel generators BUT are blocked from starting by a SI signal.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F	2	3	3	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is true of the normal plant power system other than the Safeguards Buses and the Service Building Dist System.
b	Correct- The fans are powered form the "J" MCCs that are powered from the Service Building Distribution System. The 4160V buses have an emergency power supply provided by the D3 and D4 Diesel Generators. This system is separate from the normal plant system including the Safeguards Buses.
c	Incorrect- The fans are not provided power from the Safeguards Buses so that their operation is not affected by SI signal.
d	Incorrect- The fans are not provided power from the Safeguards Buses so that their operation is not affected by a SI signal.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
027 Containment Iodine Removal System (CIRS)	K2.01	3.1	3.4

K/A Statement:	Knowledge of electrical power supplies to the following: -Fans

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguards Ventilation: Containment Air Handling System	P8180L-009H	II.C.3.c	38	1
Non-Safeguards Distr. Service Building Systems Electrical	P8186L-003B	V.D.2.a,III.C	22,11	1

Facility Learning Objective:	P8180L-009H #4.c P8186L-003B #2
Question Source:	New
Comments:	

Record #	49
RO #	40
SRO #	39

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred with degradation of core cooling
- Core cooling has been restored
- Actions of ES-1.1 "Post LOCA Cooldown And Depressurization" are in progress
- Containment pressure is 4.6 psig after peaking at 28 psig
- Containment hydrogen is reading 6.2%

Which action concerning the containment hydrogen recombiners is appropriate?

The hydrogen recombiners should...

Answers/Distracters:

a. NOT be started since the hydrogen level is in EXCESS of the detonation limit.

b. NOT be started until plant-engineering staff concurs since the hydrogen flammability limit is EXCEEDED.

c. be STARTED immediately since the hydrogen level is LESS THAN the detonation limit for adverse containment conditions.

d. be STARTED immediately since the hydrogen level is LESS THAN the flammability limit for normal containment conditions.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F,P	2	3	3	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The lower detonation limit is not exceeded.
b	Correct- For Containment pressure the use of Adverse Containment conditions should be used when pressure is above 5 psig. With normal containment conditions, the hydrogen limits for placing the recombiners in service is between 0.5% and 6.0%. If hydrogen concentration is > 6.0% [2.3%] then consultation with plant engineering staff is required. The flammability limit for hydrogen is between 4%(lower limit) and 74.2% (upper limit). The detonation limit for hydrogen in air is between 19% (lower limit and 70% (upper limit).
c	Incorrect- While the hydrogen is less than the detonation limit, the recombiner should not be started until plant engineering has reviewed the condition and determined if other action should be taken to reduce the hydrogen concentration (venting or pressurization with air).
d	Incorrect- The lower flammability limit is exceeded with potential for more than just a limited burn. The recombiner should not be started until plant engineering has reviewed the condition and determined if other action should be taken to reduce the hydrogen concentration (venting or pressurization with air).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
028 Hydrogen Recombiner and Purge Control System (HRPS)	A2.03	3.4	4.0

K/A Statement:	Ability to (a) predict the impacts of the following on the HRPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: -The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment

Reference Title:	Reference #:	Section:	Page:	Rev:
Containment Hydrogen Control	P8180L-008	III.B.1, V.C.1	10-11, 21	2
Post LOCA Cooldown And Depressurization	1ES-1.1	Step 26	17	15

Facility Learning Objective:	P8180L-008 #4
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Initial Bank question P8180L-008 002. Modified pressure and hydrogen values to give higher hydrogen limit. Changed selections to incorporate flammability limits and detonation limits for hydrogen. Correct answer changed.

Record #	50
RO #	41
SRO #	40

Stem:

(see reference)

Given the following conditions on Unit 1:

- A LOCA has occurred
- 1FR-C.1 was entered due to an inadequate core cooling condition.
- Containment pressure is 8 psig
- Containment temperature prior to the LOCA was 90 degrees F.
- Containment hydrogen concentration is 0.45%

Based on these indications, using the references provided, what is the required power setting for placing 12 hydrogen recombiner in service?

Answers/Distracters:

a. 52.0

b. 53.2

c. 57.8

d. 59.1

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-SPR	2	3	3	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This would be the correct answer if the reference power setting for 11 recombiner were used.
b	Incorrect- This would be the correct answer if the reference power setting for 11 recombiner and the 60 F temperature curve were used.
c	Correct-using the 90 F curve from fig 1 of C19.8, a Cp. factor of 1.36 is obtained. The reference power setting for 12 recombiner is 42.5. The reference power setting X Cp. = required power setting. Therefore 42.5X1.36=57.8
d	Incorrect- This would be the correct answer if the 60 F temperature curve were used.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
028 Hydrogen Recombiner and Purge Control System (HRPS)	A4.01	4.0	4.0

K/A Statement:	Ability to manually operate and/or monitor in the control room: -HRPS controls

Reference Title:	Reference #:	Section:	Page:	Rev:
Containment Hydrogen Control	P8180L-008	V.C	21	2
POST LOCA H2 ELECTRIC RECOMBINER CONTROL SYSTEM	C19.8	5.1	3,4 fig 1	10

Facility Learning Objective:	P8180L-008 #12,13
Question Source:	new
Comments:	Requires C19.8 pages 3,4 and figure 1 as references.

Record #	51
RO #	42
SRO #	41

Stem:

The plant was in the "normal electrical configuration" when it experienced an unplanned loss of 10 Bank and 1R Transformer. Assuming that all other conditions are normal, which transformer is now supplying each of the 4.16 KV safeguards buses?

Answers/Distracters:

a. Bus 15 from CT12
Bus 16 from CT12
Bus 25 from 2R
Bus 26 from 2R

b. Bus 15 from 2R
Bus 16 from CT11
Bus 25 from 2R
Bus 26 from CT11

c. Bus 15 from CT11
Bus 16 from CT11
Bus 25 from 2R
Bus 26 from CT12

d. Bus 15 from CT11
Bus 16 from CT11
Bus 25 from 2R
Bus 26 from 2R

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-I	2	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-CT 12 is unavailable with the 10 bank transformer OOS. This could be considered if CT 12 is mistakenly thought to be powered from the 161 KV bus instead of the 10 bank transformer and CT 12 aligned to unit 1.
b	Incorrect-As in d below bus 15 will not transfer to 2R and bus 26 from CT 11.
c	Incorrect-This selection could be considered if CT 12 is mistakenly thought to be powered from the 161 KV bus instead of the 10 bank transformer
d	Correct-The normal plant electrical configuration is as follows: bus 15 supplied from 1R transformer, bus 16 supplied from CT 11 transformer, bus 25 supplied from 2R transformer, bus 26 supplied from CT 12 transformer. In addition, 1 R transformer is supplied from 161 KV bus 1, which has 2 supply breakers (6H2 from the 10 bank transformer and 6H5 from the Spring Creek line). 2R transformer and CT 11 transformer are supplied from 345 KV bus 1. CT 12 transformer is supplied from the tertiary winding of the 10 bank transformer which is supplied from 345 KV bus 2. When 1R transformer is lost bus 15 transfers to is alt supply from CT 11. When the 10 bank transformer is lost the supply to CT 12 is lost, therefore bus 26 transfers to its alt supply from 2R transformer. Busses 16 and 25 are unaffected by the loss of 10 bank and CT 12 transformers except that 1 path from the grid to all of the safeguards busses is now unavailable.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
062 AC Electrical Distribution System	K4.02	3.4	3.7

K/A Statement:	Knowledge of the ac distribution system design feature(s) and/or interlock(s) which provide for the following: - Circuit Breaker automatic trips

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguards Electrical Distribution	P8186L-008	III.A,IV.E.5.b	9-10, 26-27	4
Power Sources For Safeguards Buses	B20.5	Table B20.5-2	16	3

Facility Learning Objective:	P8186L-008 #3,4,5
Question Source:	Bank
Comments:	Requal Bank question 9140L-808 #3

Record #	52
RO #	43
SRO #	42

Stem:

(see reference)

Given the following conditions for BOTH Units:

- RWST Water Cleanup is in progress for 21 RWST
- An operator is currently performing the operations to transfer SFP cooling from 121 SFP heat exchanger (HX) to 122 SFP heat exchanger (HX)
- 121 SFP HX has been isolated and 122 SFP HX placed in service
- The operator then begins to throttle open **SF-14-17,** SFP DEMIN OUT to 121 FLTR

What is the effect of this action on SFP level?

Answers/Distracters:

a. SFP level quickly rises as the content of the 21 RWST is dumped to the SFP.

b. SFP level slowly rises as the water from the 21 RWST is transferred to the SFP.

c. SFP level slowly decreases as the water from the SFP is transferred to the 21 RWST.

d. SFP level remains constant since the SFP Cooling remains isolated from the purification loop.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPR	2	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-This may be considered if the operator believes the purification loop isolation SF-14-7 is opened and the RWST head is greater than that in the SFP line. However the flow rate would be limited by capacity of the Refueling Water Purification Pump and the difference in head between that pump and the SFP Pump head.
b	Incorrect- This may be considered if the operator believes the purification loop isolation SF-14-7 is opened and the flow rate is determined to be limited by the capacity of the Refueling Water Purification Pump, and the pressure difference for that pump is greater than that in the SFP cooling loop. This does not occur.
c	Incorrect-This may be considered if the operator believes purification loop isolation SF-14-7 is opened and the SFP Cooling system pressure is greater than that of the Refueling Water Purification Pump. AT approximately 79 psig, the flow would be expected from the SFP Cooling into the Purification loop with discharge being to the 21 RWST.
d	Correct-When RWST Cleanup is initiated, manual valves SF-14-6 (SFP Purif Loop Outlet to SFP) and SF-14-7 (SFP Cooling Pumps to Purif Loop) are closed to isolate the RWST cleanup path from the SFP Cooling loop. The operation of the 122 SFP HX does not affect this alignment. The directed throttling of SF-14-17 is used to control flow through the purification loop, so if it is opened (further) the flow to/from the RWST is expected to increase within the limits of the Refueling Water Purification Pump.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
033	Spent Fuel Pool Cooling System (SFPCS)	A1.01	2.7	3.3

K/A Statement:	Ability to predict and/or monitor changes in parameters associated with operating the Spent Fuel Pool Cooling System controls including: -Spent fuel pool water level

Reference Title:	Reference #:	Section:	Page:	Rev:
Spent Fuel Pool Cooling System	P8182L-004	IV.C.4	10	1
SPENT FUEL COOLING SYSTEM	C16	5.9, 5.12	16, 18-19	33
Spent Fuel Cooling Flow Diagram	XH-1-29			Z

Facility Learning Objective:	P8182L-004 #4,5
Question Source:	New
Comments:	Flow Diagram X-HIAW-1-29 given as reference

Record #	53
RO #	
SRO #	43

Stem:

The minimum allowable Spent Fuel Pool temperature of 68°F is based on which of the following?

Answers/Distracters:

a. 68°F is the minimum analyzed temperature to ensure Adequate Shutdown Margin in the Spent Fuel Pool.

b. 68°F is the minimum allowed temperature to ensure the Spent Fuel Pool Demineralizers will remove Sulfates.

c. 68°F is the minimum analyzed temperature to ensure Brittle Fracture Prevention of the Spent Fuel Pool liner.

d. 68°F is the minimum allowed temperature to ensure the solubility of Boric Acid in the Spent Fuel Pool.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-B	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct-The limitations on the SFP include: 1- a minimum of 1800 ppm boron SHALL be maintained in the SFP, per T.S.3.8 to ensure Keff < 0.95 and 2-SFP temperature SHALL NOT be allowed to go below 68 F, the minimum analyzed temperature to ensure adequate shutdown margin.
b	Incorrect- Although the efficiency of the SFP demineralizers decreases as temperature decreases, 68 F is well within the acceptable range for the demineralizers' resin.
c	Incorrect-Brittle Fracture is a concern for all metals, but since the SFP is open to atmospheric pressure only, fracture of the liner is not a concern at 68 F. This answer could be selected if the operator thinks that the reactor vessel brittle fracture concerns apply to the SFP.
d	Incorrect-From the C12 solubility curve for Boric Acid, the concentration of acid that is required to become insoluble at 68 F would be in excess of 25000 ppm. The SFP is typically maintained at approximately 3000 ppm boron, therefore solubility of boric acid is not a limitation.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
033	Spent Fuel Pool Cooling System (SFPCS)	2.1.32	3.4	3.8

K/A Statement:	Conduct Of Operations -Ability to explain and apply all system limits and precautions.

Reference Title:	Reference #:	Section:	Page:	Rev:
Spent Fuel Pool Cooling System	P8182L-004	V.A.7.k	15	1
SPENT FUEL COOLING SYSTEM	C16	4.2	5	33

Facility Learning Objective:	P8182L-004 #4
Question Source:	New
Comments:	

Record #	54
RO #	44
SRO #	

Stem:

Given the following conditions on Unit 1:

- The Unit was taken to 33% power during a startup when a failure required the unit to be shutdown
- Current reactor power and main generator output is 18%

In this condition, the Steam Generator Level Control System is in MANUAL when the operator...

Answers/Distracters:

a. **takes** the Main Feedwater Valve controllers from AUTOMATIC to MANUAL.

b. **takes** the Bypass Feedwater Valve controllers from AUTOMATIC to MANUAL.

c. **takes** BOTH the Main Feedwater Valve controllers AND the Bypass Feedwater Valve controllers from AUTOMATIC to MANUAL.

d. **selects** Low Power Mode control, **matches** Bypass Feedwater Valve controller DEMAND and POSITION, AND then **takes** the Bypass Feedwater Valve controllers from AUTOMATIC to MANUAL.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c.	2-RI	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This condition would be true if power had been taken above 40% and the Bypass valves placed in MANUAL & Closed, as directed by procedure
b	Incorrect- This may be considered if the difference between LOW POWER/HIGH POWER control and the operation of the Main/Bypass Feed valves are misinterpreted. This also may have the operator consider whether the controller is in HIGH or LOW Power Mode.
c	Correct-The operation of FW control is power dependent (Bypass valve ops) and relationship between auto & manual operations of FW control. The feedwater control system automatically switches from Low Power Mode to High Power Mode control when both loops of feedwater control exceed 25% flow (at approximately 25% power). The Low Pressure Mode is automatically placed back in control when flow falls below 15% (approximately 15% power). The difference between Modes is the controlling signals for maintaining S/G levels (input signals) and the Mode does not directly affect valve operation in AUTO. Both the Main and Bypass Feedwater valves rare operated in AUTO and open as appropriate by the controller signals. At 40% power the operator is directed to take the Bypass Feed Valve to MANUAL and shut the valve. The auto feed control remains for the Main Feed valve. The feedwater control system is not considered to be in MANUAL until BOTH the Main Feedwater controller and the Bypass Feedwater Controller are taken to MANUAL. Bumpless transfer occurs in all conditions, since for both controllers the auto track. In this case, initially both the Main & the Bypass Feed Valve controllers are in AUTO. Both must be taken to MANUAL to switch the system to MANUAL control
d	Incorrect- This addresses automatic transfer control (bumpless) and the LOW POWER/HIGH POWER Mode of control. Switching between Modes is also automatic.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
035	Steam Generator System (S/GS)	A4.01	3.7	3.6

K/A Statement:	Ability to manually operate and/or monitor in the control room: -Shift of S/G controls between manual and automatic control, by bumpless transfer

Reference Title:	Reference #:	Section:	Page:	Rev:
Steam Generator Level Control System	P8174L-006	II.B , III.C, IV.C.1	8-9, 13-14, 27	3
REACTOR CONTROL SYSTEM	B7	3.5.2, 3.5.3, 4.5.1	26-29, 38	4
UNIT 1 POWER OPERATION	1C1.4	5.1.14, 5.1.19	14,18	24

Facility Learning Objective:	P8174L-006 #4,5
Question Source:	Significantly modified from NRC Exam Bank
Comments:	Salem 2/99 NRC exam.

Record #	55
RO #	45
SRO #	44

Stem:

Given the following conditions:

- The plant is at 70% power
- Failure of automatic control results in one Main Feedwater valve going closed.
- The operator takes manual control and rapidly opens the valve to near its previous position.

Which of the following is an INITIAL response to re-opening the valve?

Answers/Distracters:

a. Turbine power output INCREASES due to increase in steam temperature.

b. Pressurizer level INCREASES due to increase in RCS Tavg.

c. S/G level SWELLS due to the rapid addition of feedwater.

d. Rods STEP IN due to the increase in reactor power.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	3-PEO	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect—Turbine power is not changed by the event
b	Incorrect-RCS Tavg is expected to decrease due to the lower cold leg temperature. However since the other loop is not affected and auctioneered Tave is used for the Pressurizer reference level, the level setpoint is not expected to change
c	Incorrect- The rapid addition of the colder feed will result in collapse of the bubbles in S/G boiling mixture, the indicated level will decrease with this collapse, resulting in shrink not swell
d	Correct-The rapid addition of cooler feedwater, resulting in cooler primary cold leg water temperature will result in an increase in power output by the core. The increase in NIS output measured against the constant turbine power will result in a rate of-change difference for the rod control system. With this the rods will step in to attempt to lower reactor power

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
035	Steam Generator System (S/GS)	K5.01	3.4	3.9

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to the S/GS: -Effect of secondary parameters, pressure, and temperature on reactivity

Reference Title:	Reference #:	Section:	Page:	Rev:
Rod Control & Rod Position Indication	P8184L-005	IV.C.4	21-23	2
Secondary Accidents	P8161L-006	V.	15-16	2

Facility Learning Objective:	P8184L-005 #1 P8161L-006 #5
Question Source:	Significantly modified from NRC Exam Bank
Comments:	Kewaunee 12/2000 NRC exam. Changed conditions in 3 selections to qualify response of auto rods to NIS response. Correct answer changed.

Record #	56
RO #	46
SRO #	

Stem:

The following conditions exist on unit one:

- SG "A" steam flow ➡ 1,000 lbm/hr
- SG "B" steam flow ➡ 800,000 lbm/hr
- SG "A" level ➡ 60% NR
- SG "B" level ➡ 10% NR
- TD AFW pump ➡ running
- MD AFW pump ➡ running
- RCS Tavg ➡ 520 F
- Containment Pressure ➡ 8 psig

If NO operator action has been taken, which of the following indicates the expected status of the main steam isolation valves?

	"A" MSIV	"B" MSIV

Answers/Distracters:

	"A" MSIV	"B" MSIV
a.	OPEN	OPEN
b.	OPEN	SHUT
c.	SHUT	OPEN
d.	SHUT	SHUT

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-F	2	4	2	2	Prairie Island	9/10/01

Basis for answers:

a	Incorrect – A MSIV closure signal exists for "B" loop.
b	Correct - A Steam Line Isolation Signal is Initiated by any of the following: (1) 2/3 Cont. pressure >17 psig (closes both MSIVs) (2) The combination of 1/2 HI-HI Stm Flow (4.47E6 #/hr) and an "S" Signal (only the affected loop's MSIV) (3) The combination of 1/2 HI Stm flow (.745E6 #/hr + 2/4 Tavg <540°F + "S" Signal (only the affect loop's MSIV) SI has actuated on high containment pressure. Only "B" loop meets isolation criteria based on item (3)
c	Incorrect - The opposite condition is expected for each MSIV
d	Incorrect - The containment pressure is below the value that would automatically isolate both SGs.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
039	Main and Reheat Steam System	A3.02	3.1	3.5

K/A Statement:	Ability to monitor automatic operations of the MRSS including: Isolation of the MRSS

Reference Title:	Reference #:	Section:	Page:	Rev:
Engineered Safeguards System	P8180L-006	V.B.6.a	21	3
Logic Diagrams - Safeguards Actuation Signals	X-HIAW-1-242			D

Facility Learning Objective:	P8180L-006 #4f
Question Source:	Facility requal bank question P8180L-006 005
Comments:	

Record #	57
RO #	
SRO #	45

Stem:

(see reference)

Which of the following alarms, actuated by a single valid input condition, would require the earliest shutdown of the reactor if that condition continues to exist? **(Assume conditions do not result in an automatic reactor trip.)**

Answers/Distracters:

a. **47010- 0105**, 11TD AFWP ACCUMULATOR LO AIR PRESSURE

b. **47010-0305**, 11TD AWFP LOCAL CONTROL SI AUTO START BLOCKED

c. **47011-0205**, 11 OR 12 MAIN STEAM RELIEF VALVE LOCAL CONTROL

d. **47011-0505**, 11 OR 12 STM GEN ISOLATION VALVE LO AIR PRESS

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-SPK	2	4	2	2	Prairie Island	9/10/01

Basis for answers:

a	Incorrect - This indicates possible failure of air supply to TD AFW pump start valve (CV-31998). This valve fails open on loss of air starting the TD AFWP. TD AFWP operability is not affected.
b	Incorrect - This indicates an LCO condition for the TD AFW, but TS 3.4.B.2 applies allowing for pump inoperability for 72 hours without shutdown required.
c	Correct - PORV is operable when ability to operate remotely in auto or manual and block valve open. With alarm 47010-0105 in, either 11 or 12 PORV control is in LOCAL (control at Hot Shutdown Panel). Therefore the valve is inoperable. TS 3.4.A.2 is applicable in that the POWER OPERATION may continue for 48 hours.
d	Incorrect - Indicates low air pressure to MSIV operator. The Trip Valve is Air Operated to Open. The alarm indicates that MSIV will go shut and not reopen if air pressure continues to decrease. This would result in a turbine trip-reactor trip if the MSIV comes off its OPEN position. Since no auto actions occur, the MSIV remains open. This condition does not affect operability of MSIV since each valve closes as result of actuation Two redundant, parallel, air cylinder vent solenoids to vent air away from the operators.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
039	Main and Reheat Steam System	2.4.45	3.3	3.6

K/A Statement: Ability to prioritize and interpret the significance of each annunciator or alarm.

Reference Title:	Reference #:	Section:	Page:	Rev:
Main and Auxiliary Steam System	P8174L-001	VI.D.2, XV.D.2	16, 34-35	2
Alarm Response Procedures	C47011	47011-0205	1	15
Prairie Island Technical Specifications		3.4.A.2	3.4-1	123

Facility Learning Objective:	P8174L-001 #5,8
Question Source:	New
Comments:	Provide T.S.3.4 as available reference

Record #	58
RO #	47
SRO #	46

Stem:

Given the following conditions on Unit 1:

- The reactor has been shutdown from 100% power
- Steam Dumps have been placed in the "STEAM PRESSURE" mode
- The Shift Supervisor has directed that RCS Tavg be controlled 4°F above the LO-LO TAVG STEAM DUMP INTERLOCK setpoint prior to initiating a cooldown

What value would be required to be set on the MAIN STM HDR PRESS controller in auto to maintain Tavg?

Answers/Distracters:

a. 71.8%

b. 70.0%

c. 67.7%

d. 65.4%

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	3-SPR	2	4	3	3	Prairie Island	9/10/01

Basis for answers:

a	Incorrect – This is the normal no-load pressure setpoint for maintaining 547°F RCS temperature.
b	Correct - At no-load conditions the MAIN STM HDR PRESS controller is set at the value equivalent to the saturation pressure to maintain Tavg (0°F delta-T). The LO-LO TAVG STEAM DUMP INTERLOCK setpoint is 540°F. The desired temperature would then be 544°F. The saturation pressure for 544°F is 980.2 psig as determined from steam tables.
c	Incorrect-This is the saturation pressure setpoint associated with 540°F
d	Incorrect-This is the saturation pressure setpoint for 536°F, which is 4°F BELOW the LO-LO TAVG STEAM DUMP INTERLOCK setpoint.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
041 Steam Dump System and Turbine Bypass Control	K5.02	2.5	2.0

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to the SDS:
- Use of steam tables for saturation temperature and pressure

Reference Title:	Reference #:	Section:	Page:	Rev:
Steam Dump Control System	P8174L-002	IV.B.1, 3	14-15, 16-17	3
Steam Tables				

Facility Learning Objective:	P8174L-002 #4,7
Question Source:	New
Comments:	Steam tables and calculator required

Record #	59
RO #	48
SRO #	47

Stem:

Given the following conditions on Unit 1:

- The reactor is at 75% steady state power.
- All systems are in automatic control.
- Main turbine control is in "IMP IN" with the valve position limiter set at 95%.

Under these conditions, what would be the response if the condenser steam dump "**CV-31100**" failed OPEN due to a valve regulator failure?

Answers/Distracters:

a. Turbine load **decreases** by 5% AND reactor power remains **constant**. The operator can restore conditions by taking either Bypass Interlock switch to "OFF/RESET".

b. Turbine load remains **constant** AND reactor power **increases** by 5%. The operator can restore conditions by taking the Steam Dump Mode Selector switch to "STEAM PRESSURE".

c. Turbine load **decreases** by 7.5% AND reactor power remains **constant**. The operator can restore conditions by taking the Steam Dump Mode Selector switch to "STEAM PRESSURE".

d. Turbine load remains **constant** AND reactor power **increases** by 7.5%. The operator can restore conditions by taking either Bypass Interlock switch to "OFF/RESET".

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	2-RI	2	4	3	3	Prairie Island	9/10/01

Basis for answers:

a	Incorrect - Turbine load will not decrease in IMP IN. (But may be expected to in IMP OUT)
b	Incorrect - Taking the Steam Dump Selector to STEAM PRESSURE will not stop airflow to the regulator, but is normally selected if failure of RCS Temperature or Turbine First Stage pressure occurs. (that may result in arming or opening of dump valves)..
c	Incorrect - Turbine load will not decrease in IMP IN. (But may be expected to in IMP OUT) and taking the Steam Dump Selector to STEAM PRESSURE will not stop airflow to the regulator, but is normally selected if failure of RCS Temperature or Turbine First Stage pressure occurs (that may result in arming or opening of dump valves)..
d	Correct - With IMP IN at 95%, the turbine will respond to changes in turbine first stage pressure up to 95% of the 100% turbine first stage impulse pressure (520 psig) to position the control valves, approaching a constant MWe control. Thus turbine load is not expected to change. The steam dump valve is rated at 7.5% total steam flow (100% power). This is equivalent to 7.5% power change. Since turbine load remains the same, the reduction in Tcold will raise reactor power by the amount equivalent to the additional steam flow. Since the failure involves the regulator air supply to the valve can be removed by operating either of the two Bypass Interlock switches to vent air from the supply line to the regulator.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
045	Main Turbine Generator System	A2.08	2.8	3.1

K/A Statement: Ability to (a) predict the impacts of the following on the MT/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
- Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)

Reference Title:	Reference #:	Section:	Page:	Rev:
Steam Dump Control System	P8174L-002	III.B.2, V.A.2.e	9-10, 20	3
Turbine Control	P8176L-001	IV.D.2.d)	48-49	2

Facility Learning Objective:	P8174L-002 #5, 9; P8176L-001 #2a
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8174L-001 024. Changes include failure of Condenser Dump Valve compared to S/G PORV. Dump valve has higher release value (7.5% vs. 5%). Also added conditions that will mitigate the event.

Record #	60
RO #	
SRO #	48

Stem:

Given the following conditions on Unit 1:

- The Unit is at 80% power
- Steam Dump control is in Tavg mode
- The Main Steam Header Pressure Controller is in MANUAL AND set to ZERO

During a subsequent 100 MW load rejection, the RO reports that the red channel cold leg RTD FAILED HIGH AND the following conditions are noted for the Steam Dump System:

- Aqua Status Light **47014-0405,** LOSS OF LOAD INTERLOCK, is lit
- Aqua Status Light **47014-0502,** CDSR STEAM DUMP PREMISSIVE, is lit

What action should be taken with respect to the Steam Dumps to restore the system to normal conditions (ONCE CORRECTIVE ACTIONS ARE COMPLETED for the failed Tavg channel)?

Answers/Distracters:

a. Momentarily **place** STEAM DUMP MODE selector to RESET.

b. Momentarily **place** ONLY the STEAM DUMP CDSR VLV LO-LO TAVG INTLK B-P TRAIN A switch to OFF/RESET.

c. Momentarily **place** BOTH STEAM DUMP CDSR VLV LO-LO TAVG INTLK B-P TRAIN A and TRAIN B switches to BYPASS INTERLOCK.

d. **Place** the STEAM DUMP MODE selector to STEAM PRESS.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-DR,RI	2	3	3	3	Prairie Island	9/10/01

Basis for answers:

a	Correct- The Status light 0502 indicates normal operation status for the steam dump system, with the permissives for opening the condenser steam dump is present. A failure of the RTD in conjunction with a greater than 10% load reduction results in actuation of the loss of load interlock providing the other necessary inputs for opening all steam dumps (loss of load as indicated by 0405 light and Tavg higher than Tref with RTD failure). If no action is taken the steam dumps would remain open until the LO-LO Tavg interlock is actuated (540°F). The steam dump may be returned to normal closed condition by taking the control to RESET to clear the loss of load signal.
b & c	Incorrect- This can be considered since the switches can individually and as the pair affect steam dump operation. The identifying condition however is the light indicating a loss of load condition (with Tavg-Tref mismatch) opening the steam dumps. Taking the one or both controls to OFF/ REST or BYPASS INTERLOCK will have no permanent effect on the dumps. Placing A in OFF Reset will close the steam dump valves but they would reopen as soon as they are restored to ON. Also this does not restore Steam Dumps to "normal" since they would not open for any actuation signal. Taking both switches to BYPASS RESET has no effect as long as Tavg is higher than 540°F. As before steam dumps remain open.
d	Incorrect- This can be considered because for some failure (Tref) it is desired to place the Steam Dump control to Steam Pressure mode. This action however does not have the desired affect since the (pressure) Controller is in manual. Steam Dumps would not open for any actuation condition (high pressure).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
041	Steam Dump System (SDS) and Turbine Bypass Control	2.4.45	3.3	3.6

K/A Statement: Emergency Procedures / Plan
-Ability to prioritize and interpret the significance of each annunciator or alarm

Reference Title:	Reference #:	Section:	Page:	Rev:
Steam Dump Control System	P8174L-002	III.E.2 IV.B.2	11,15-16	3
Alarm Response Procedure	C47014-0405	Subs. Action	1	12
Interlock Logic Diagram Main Steam System	NF-40322-2			R

Facility Learning Objective:	P8174L-002 #4,5
Question Source:	New
Comments:	Replaces 045 2.1.14 (SRO). No 045 SYSTEM KA at SRO-only level selected on replacement outline. KA selected from potential SRO-only related KA on replacement outline.

Record #	61
RO #	49
SRO #	

Stem:

Given the following conditions on Unit 1:

- Reactor power is 90%
- ALL Heater Drain Pump Motors have uncoupled due to a fault in the HD Pump Control Cabinet.
- The following annunciators are in alarm:
  47003-0403, CONDENSATE BYPASS TO HEATER DRAIN PUMP OPEN
  47010-0602, COND B-P FEEDWATER PUMP OPEN
- Feedwater pumps suction pressure has stabilized at 250 psig

What Operator action should be taken?

Answers/Distracters:

a. **Verify** THREE Condensate Pumps are running.

b. **Open CV-31122**, CONDENSATE RECIRCULATION SPRAY VALVE.

c. **Verify CV-31040**, 11 Heater Drain Tank Bypass to 1A Condenser, is CLOSED.

d. **Reduce** turbine load to less than 60% AND **stop** ONE Feedwater Pump.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-P	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - The alarms actuate when feedwater suction pressure falls below 220 psig. Three automatic actions take place: 1) CV-31039 opens, 2) CV-31087 Condensate Bypass to Feedwater pumps opens and the standby Condensate pump starts. The operator should ensure all automatic actions have occurred, including the start of the standby Cond Pump, meaning three Cond Pumps running.
b	Incorrect - CV-31122 opens on low condensate flow (at 3000 gpm decreasing) to ensure adequate cooling flow through Cond Pumps (and closes at 6500 gpm increasing flow).
c	Incorrect – This is the correct action for low level in the Heater Drain Tank. Heater Drain Tank level is expected to go high and cv-31040 should open.
d	Incorrect - With the reactor power less than rated thermal power, turbine load reduction/MFW Pump stopping is not required as long as adequate FW Pumps suction pressure (and SG levels) are maintained. This is indicated by pressure stabilizing at 250 psig.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
056	Condensate System	2.4.50	3.3	3.3

K/A Statement: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Reference Title:	Reference #:	Section:	Page:	Rev:
Condensate and Feedwater	P8174L-003	III.D.2.a, 8.e	19, 24	4
Alarm Response Procedure	C47003	47003-0403	1	5
Alarm Response Procedure	C47010	47010-0602	1	25

Facility Learning Objective:	P8174L-003 #4, 6
Question Source:	NRC Exam Bank
Comments:	Kewaunee 12/2000 NRC Exam. Changed conditions in premise to match specific plant indication and values (Alarms). Changed three selections (including correct answer) based on plant specific response to conditions.

Record #	62
RO #	50
SRO #	49

Stem:

Given the following conditions on Unit 2:

- The reactor has tripped from 50% power
- RCS Tavg is 550°F and stable
- RCS Press is 2235 psig and stable
- Both S/G Press are 1010 psig and stable
- 21 S/G level is 25% narrow range
- 22 S/G level is 36% narrow range

Which of the following conditions describes the expected condition of the Feedwater Regulating Valves (FRV) AND the demand position on its associated controller?

	21 FRV POSITION	21 FRV DEMAND	22 FRV POSITION	22 FRV DEMAND

Answers/Distracters:

a.	OPEN	OPEN	CLOSED	CLOSED
b.	CLOSED	OPEN	CLOSED	OPEN
c.	CLOSED	OPEN	CLOSED	CLOSED
d.	CLOSED	CLOSED	CLOSED	CLOSED

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-DR	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect- This could be considered if this were the Bypass valves operating in AUTO. Only the 11 B/P FRV would be open with demand open for its controller. 12 B/P FRV would be closed and demanded closed since its level is above the setpoint.
b	Incorrect- This would be the expected post trip condition for the FRVs when both SG levels are below the reference level.
c	Correct-With a reactor trip signal P-4 signal present and RCS Tavg below the Lo Tavg setpoint (554°F), the FRVs will be closed. However, with SG level below the level setpoint, the DEMAND signal from the controller will provide a position signal of OPEN for its associated valve. At 0% power the S/G programmed level is 33%. So both 11 and 12 FRVs would be closed. Since the level in 11 SG is below setpoint, its controller demand would be open, and since the level in 12 SG is above the setpoint, its controller demand would be closed.
d	Incorrect- This may be considered since the signal provided closes the FRVs. However the demand signal on the controller remains directly unaffected by the closure signal and is dependent on SG level (error).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
059	Main Feedwater (MFW) System	A4.08	3.0	2.9

K/A Statement:	Ability to manually operate and/or monitor in the control room: - Feed regulating valve controller

Reference Title:	Reference #:	Section:	Page:	Rev:
Steam Generator Level Control System	P8174L-006	III.A.2, IV.D.2.d	10, 28-29	3
Feedwater Control & Isolation Logic	X-HIAW-1-244			D
ADFCS	X-HIAW-1-249-2&3			A

Facility Learning Objective:	P8174L-006 #1,3,6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8174L-006 011. Modified to give condition of FRV only and to identify Controller demand signal associated with the conditions. B/P FRV positions are removed from the question.

Record #	63
RO #	51
SRO #	50

Stem:

Given the following conditions on Unit 1:

- Reactor power is 7%
- Steam dump to the condenser is OPEN maintaining Steam Header Pressure
- Turbine is latched and rolling at 1800 RPM
- 11 Main Feedwater Pump is running
- All Condensate Pumps have just TRIPPED due to low water level in the hotwell

What will occur in the SG feed systems?

Answers/Distracters:

a. 11 Main Feedwater Pump **trips** IMMEDIATELY; the AFW pumps **start** WHEN the Main Feedwater pump trips.

b. 11 Main Feedwater Pump **trips** IMMEDIATELY; the AFW pumps **start** WHEN SG level reaches the Lo-Lo setpoint.

c. 11 Main Feedwater Pump **trips** after a 15-second time delay; the AFW pumps **start** WHEN the Main Feedwater pump trips.

d. 11 Main Feedwater Pump **trips** after a 15-second time delay; the AFW pumps **start** WHEN SG level reaches the Lo-Lo setpoint.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-F	2	2	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - Before exceeding 2% power, The main feedwater system is aligned for maintaining SG levels. The AFW system is shutdown and aligned for (automatic) At-Power operations by placing the controls in AUTO. The Main FW Pump(s) trip whenever a loss of required number of Condensate Pumps occurs (1/1 logic). At least ONE Cond Pump must be running for any Main FW Pump to continue running. The AFW Pumps, 11 & 12, start whenever the 11 & 12 (or 21 & 22) FW pump breakers trip with AFW selector switch in AUTO
b	Incorrect-While the Lo-Lo SG level will auto start the AFW Pumps, this will take some time for the level to drop to this point. The AFW pumps will have already started on the trip of the Main FW Pumps.
c	Incorrect -There is a 5-sec delay associate with the low suction pressure trip for the Main FW Pumps, suction pressure falls below 200 psig for at least 5 seconds.
d	Incorrect -While the Lo-Lo SG level will auto start the AFW Pumps, this will take some time for the level to drop to this point. The AFW pumps will have already started on the trip of the Main FW Pumps. Additionally, there is a 5-sec delay associate with the low suction pressure trip for the Main FW Pumps, suction pressure falls below 200 psig for at least 5 seconds.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
059	Main Feedwater (MFW) System	K3.02	3.6	3.7

K/A Statement: Knowledge of the effect that a loss or malfunction of the MFW System will have on the following:
- AFW System

Reference Title:	Reference #:	Section:	Page:	Rev:
Condensate and Feedwater	P8174L-003	III.D.8.b	22-23	4
Auxiliary Feedwater System	P8180L-007	IV.A.2, IV.B.1	18, 21-22	5
UNIT 1 STARTUP PROCEDURE	1C1.2	5.10.7	68	24

Facility Learning Objective:	P8174L-003 #6; P8180L-007 #4
Question Source:	Facility requal exam bank question P8174L-003 005
Comments:	Minor wording changes made to premise and question lead-in.

Record #	64
RO #	52
SRO #	51

Stem:

Given the following conditions for both Units:

- Unit 1 is at 5% power
- Unit 2 is at 100% power
- 12 MD AFW Pump has indications of steam binding AND is isolated
- 11 TD AFW Pump failed its surveillance AND was declared inoperable
- 21 MD AFW discharge was cross-connected to Unit 1
- Following the cross-tie, 21 SG level falls to 10% Narrow Range due to a feedwater valve problem

What is the response of the AFW System?

AFW flow is automatically initiated to Unit 2 SGs from...

Answers/Distracters:

a. 22 AFW Pump only AND will indicate greater than 100 gpm to each Unit 2 SG.

b. 22 AFW Pump only AND will indicate less than 100 gpm to each Unit 2 SG.

c. BOTH 21 and 22 AFW Pumps AND will indicate greater than 100 gpm to each Unit 2 SG.

d. BOTH 21 and 22 AFW Pumps AND will indicate less than 100 gpm to each Unit 2 SG.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	3-PEO	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct-With the 21 MD AFW Pump discharge cross-tied to Unit 1, the 21 Pump is maintained in MANUAL with a dedicated operator assigned for controlling the Pump. The TD AFW Pumps for each Unit should remain in AUTO operation. With SG level less than 13%, an AFW start signal is generated and the TD AFW Pump for the Unit with the affected SG(s) will start, and flow will go to the SGs. (NOTE also that a reactor trip signal is also generated by this condition). Additionally, the design of the pumps is to maintain greater than 200 gpm total per unit with one pump OOS.
b	Incorrect -This may be a misconception since only one pump is aligned to unit 2 and flow will be less than normally observed, however it will not be below 100 gpm to each SG.
c	Incorrect –This may be a misconception if the x-tied lineup is not fully understood but the design flow is understood.
d	Incorrect – This may be a misconception if the x-tied lineup is not fully understood and they believe it would result in degraded flow to unit 2 SG's.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
061 Auxiliary / Emergency Feedwater System	A1.03	3.1	3.6

K/A Statement:	Ability to predict and/or monitor changes in parameters associated with operating the AFW System controls including: - Interactions when multi unit systems are cross tied

Reference Title:	Reference #:	Section:	Page:	Rev:
Auxiliary Feedwater System	P8180L-007	V.D.2	33	5
AUXILIARY FEEDWATER SYSTEM UNIT 1	1C28.1	5.7.2	16	5

Facility Learning Objective:	P8180L-007 #7
Question Source:	new
Comments:	

Record #	65
RO #	53
SRO #	

Stem:

Given the following conditions:

- An electrical fire has occurred in MCC 2TA1 requiring deenergization of MCC 2TA1.
- The Fuel Oil Transfer Pump Selector Switch on the D5 Benchboard is in the 21-pump position.
- The fuel oil outlet line has developed a leak at the bottom of the D5 fuel oil Day Tank.

Which of the following will occur in regard to the D5 fuel oil system?

Answers/Distracters:

a. LOW level in the D5 fuel oil Day Tank will start ONLY 21 D5 Fuel Oil Pump in attempt to refill the Day Tank.

b. LOW-LOW level in the D5 fuel oil Day Tank will start ONLY 23 D5 Fuel Oil Pump in attempt to refill the Day Tank.

c. LOW-LOW level in the D5 fuel oil Day Tank will start BOTH 21 and 23 D5 Fuel Oil Pumps in attempt to refill the Day Tank.

d. NIETHER 21 or 23 D5 Fuel Oil Pumps will start AND the D5 fuel oil Day Tank will continue to drain.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-PEO	2	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-This answer would be correct as described in d below if power were available to 21 fuel oil pump.
b	Incorrect-This could be considered if the candidate thinks 21 and 23 fuel oil pumps come from different power supplies.
c	Incorrect-This would be a correct response if the leak were large enough to exceed the capacity of one fuel oil pump and power was available to both fuel oil transfer pumps.
d	Correct-As level in the D5 Day Tank decreases the low-level setpoint is reached. The Fuel Oil Pump selected on the Fuel Oil Transfer Pump Selector Switch would start to refill the day tank. The deenergized MCC is the power supply to both 21 an 23 fuel oil transfer pumps. 21 pump from 2TA1 cell A2 and 23 pump from 2TA1 cell B2. Therefore, 21 pump will not start. When the low-low-level setpoint is reached the back-up pump, 23, would start but since it also is powered from the deenergized MCC neither pump starts. The level continues to drop untill the leak is uncovered.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
064 Emergency Diesel Generator System	K2.02	2.8	3.1

K/A Statement:	Knowledge of electrical power supplies to the following: - Fuel oil pumps

Reference Title:	Reference #:	Section:	Page:	Rev:
D5/D6 Generator set/auxiliaries	8186L-014	V.B&C	17	1
Interlock logic diagram D5/D6 fuel oil system	NF-118845			D

Facility Learning Objective:	8186L-014 #3
Question Source:	new
Comments:	

Record #	66
RO #	54
SRO #	52

Stem:

(see reference)

D1 diesel generator is loaded onto its respective bus for testing following an overhaul. The following conditions were just established.

- Generator power is 3100 KW
- Reactive load is 1400 KVAR delivered

Which of the following is the LONGEST amount of time the generator can remain at the above conditions without exceeding the machine ratings?

Answers/Distracters:

a. 0.25 hours

b. 0.75 hours

c. 900 hours

d. 1100 hours

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	3-SPR	2	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct-The engine must be overhauled if in operation between 3000-3250 KW is 0.5 hours.
b	Incorrect- The engine must be overhauled if in operation between 3000-3250 KW is 0.5 hours, therefore this answer would result in exceeding the ½ hour rating
c	Incorrect-This answer is plausible in that it is less than 1000 hours. The engine has a 1000 hour rating of 2750-3000 KW.
d	Incorrect-This answer is plausible in that it is greater than 1000 hours. The engine has a 1000 hour rating of 2750-3000 KW.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
064	Emergency Diesel Generator (ED/G) System	K4.04	3.1	3.7

K/A Statement:	
	Knowledge of ED/G System design feature(s) and/or interlock(s) which provide for the following: -Overload ratings

Reference Title:	Reference #:	Section:	Page:	Rev:
D1/D2 Diesel Generators	1C20.7	Figure 1	74	13
Diesel Generators lesson plan	P8186L-004	III.A	11	3

Facility Learning Objective:	P8186L-004 #6
Question Source:	1997 Prairie Island NRC exam.
Comments:	

Record #	67
RO #	55
SRO #	53

Stem:

The following is a timeline of activities associated with the 121 ADT Monitor Tank:

- 1000 - 121 ADT Monitor Tank is placed on recirc
- 1200 - 121 ADT Monitor Tank is sampled
- 1230 - Gen. Supt. Radiation Protection & Chemistry authorizes the release.
- 1315 - Shift Supervisor approves Discharge Permit
- 1700 - Shift turnover
- 1910 - Commenced release of 121 ADT Monitor Tank to the river

What is the problem associated with these actions AND what action should be taken once the problem is identified?

Answers/Distracters:

a. The current Shift Supervisor did NOT approve the release. **Stop** the release until the current Shift Supervisor has signed the Discharge Permit.

b. The Chemistry sample was NOT representative of the Tank contents. **Stop** the release AND **place** 121 ADT Monitor Tank on recirc.

c. Too much time has elapsed between approval of the Permit and initiation of the discharge. **Stop** the release AND **reprocess** 121 ADT Monitor Tank.

d. Discharging directly from 121 ADT Monitor Tank is NOT allowed. **Stop** the release AND **transfer** the contents of the tank to 121 CVCS Monitor Tank for release.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect-There is no requirement for the current Shift Supervisor to re-approve release. Planned releases should have been covered during turnover.
b	Correct – In inadequate recirc was done on the tank prior to release. C21.1 requires the tank be circulated for 4 hours prior to sampling.
c	Incorrect-There is no time limit for initiation of release following proper approval of release, provided tank conditions have not changed (additions).
d	Incorrect-While release may be made from CVCS Monitor Tank, the tanks are separate and used for different purposes.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
068	Liquid Radwaste System	A2.02	2.7	2.8

K/A Statement:	Ability to (a) predict the impacts of the following on the Liquid Radwaste System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: - Lack of tank recirculation prior to release

Reference Title:	Reference #:	Section:	Page:	Rev:
Radioactive Waste Liquid	P8182L-001A	IV.G.5.b, VII.A	37, 42	2
Releasing No. 121 ADT Monitor Tank To The River	C21.1-5.1	5.2.1	6	21
OFFSITE DOSE CALCULATION MANUAL (ODCM)	H4	4.2.2 Table 2.1	40-44, 81-82	15

Facility Learning Objective:	P8182L-001A #4
Question Source:	New
Comments:	Related to LER 92-16 "Release of ADT w/o sampling"

Record #	**68**
RO #	**56**
SRO #	**54**

Stem:

Which of the following will automatically CLOSE **CV-31414**, 11 Steam Generator Blowdown (SGB) Control Valve?

Answers/Distracters:

a. LOW level in the SGB Flash Tank.

b. HIGH failure of Radiation Monitor 1R-19.

c. HIGH temperature on the outlet of the SGB Flash Tank.

d. TRIP of either the 11 MD or the 12 TD Auxiliary Feedwater pump.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
b	1-I	2	2	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect- High level in the SDB flash tank results in valve closure. The valves automatically reopen when the high level condition clears.
b	Correct-A SG high activity signal from Radiation Monitoring System (RMS) channel R-19 will result in closure of 11 and 12 SGBD to Flash Tank flow control valves (CV-31414 & CV-31415). Other signals will also close these valves including: AFW Pump start, High-high temperature from SGBD HX outlet, Flash Tank high level
c	Incorrect-- The high-high temperature is sensed at the outlet of the SGBD HX (just prior to 1R19 detector).
d	Incorrect- Start of the AFW Pump(s), not trip, will result in closure of the valves. Also note that under normal operating conditions, trip of both Main FW Pumps results in auto-start of AFW Pump.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
068 Liquid Radwaste System	K6.10	2.5	2.9

K/A Statement:	Knowledge of the effect of a loss or malfunction of the following will have on the Liquid Radwaste System: - Radiation monitors

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Radioactive Waste Liquid	P8182L-001A	III.A.3.b	8	2
Radiation Monitoring System	P8182L-002	V.D.4.d	28	4
Interlock Logic Diagram Steam Gen. Blowdown Unit 1	NF-40331-1			W

Facility Learning Objective:	P8182L-001A #5 P8182L-002 #5
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8182L-002 006.

Record #	69
RO #	57
SRO #	55

Stem:

Given the following conditions:

- Both units are at 100% power
- RCS activity is elevated in unit 2 due to a pin-hole fuel leak
- The following activities are occurring in the Aux Building:
    - Fuel handling in the Spent Fuel Pool for top nozzle inspections
    - Replacement of 22 Seal Water Return filter.
    - TN-40 Cask decon and drying activities in the Cask Decon area
    - Transfer of water from the ADT Collection Tanks to the ADT Condensate Receiver Tanks
- An automatic actuation of 122 Aux Building Special Ventilation has occurred

Which of the following events has caused the actuation of 122 Aux Building Special Exhaust?

Answers/Distracters:

a. DAMAGE to a spent fuel assembly due to failure of the handling tool.

b. FILLING and VENTING of 22 Seal Water Return Filter.

c. FAILURE of the TN-40 Cask Vacuum Drying System vacuum hose.

d. FAILURE of the ADT Collection Tank Pump seal.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	2-RI	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect – A failed fuel assembly in the SFP is expected to result in an actuation of SFP Special Vent but not ABSV
b	Correct - 22 Seal water return filter is located in the filter room. The filter room ventilation system exhaust is directed into the Unit 2 Aux Bldg. Exhaust stack. Venting of this filter with elevated RCS activity levels is expected to result in actuation of ABSV (2R-30)
c	Incorrect – The Cask Decon area is outside of the Aux Bldg Special Vent Zone and the TN-40 vacuum drying system discharge is aligned to the SFP room.
d	Incorrect – All piping and components associated with this transfer are located in the Rad Waste Bldg and protected by the Rad Waste Bldg ventilation system.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
071 Waste Gas Disposal System	A1.06	2.5	2.8

K/A Statement:	Ability to predict and/or monitor changes in parameters associated with operating the Waste Gas Disposal System controls including: - Ventilation system

Reference Title:	Reference #:	Section:	Page:	Rev:
Aux Bldg Special Vent System	P8180-009B	IV.B.1	11	1
Flow Diagram -- Auxiliary Building Special Vent	NF-39600			AS

Facility Learning Objective:	P8180-009B #4
Question Source:	New
Comments:	Related to actual plant event (Internal Operating Experience Assessment (ERTF) report 94-17 "Unplanned Actuation of ABSV System")

Record #	70
RO #	58
SRO #	56

Stem:

As pressure in the Low Level Waste Gas common vent header INCREASES, which of the following auto actions occur to prevent excess pressure?

Answers/Distracters:

a. The hydrogen recombiner inlet CLOSES at 2.7 psig.

b. The backup waste gas compressor STOPS at 2.3 psig.

c. The Low Level GDT to CVCS HUT pressure control valve **CV-31272** OPENS fully at 2.5 psig.

d. The waste gas compressor discharge is directed to the standby gas decay tank at 3.0 psig.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-I	2	4	1	1	Prairie Island	9/10/01

Basis for answers:

a	Correct - For the vent header, vent header pressure maintained between 0.65 psig and 2.7 psig (normally between 1.5 and 2.0 psig). On an increasing pressure transient (ie CVCS HUT Levels going up) automatic control actions are: a) First - CV-31272 goes fully closed; b) Second - the second WGC starts at ~ 2.3 psig (Auto) WGC will auto stop at ~ 1.5 psig; c) Third - H2 Recombiner Inlet closes at ~ 2.7 psig (reset at 1.9 psig).
b	Incorrect - As noted, the backup WGC starts automatically and takes suction on the vent header to reduce vent header pressure.
c	Incorrect - CV-31272 goes fully closed as vent header pressure increases to prevent pressurization of CVCS HUT.
d	Incorrect - The WGC discharge is directed to the standby GDT if on service GDT pressure exceeds 100 psig.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
071	Waste Gas Disposal System	A3.02	2.8	2.8

K/A Statement:	Ability to monitor automatic operations of the Waste Gas Disposal System including: - Pressure-regulating system for waste gas vent header

Reference Title:	Reference #:	Section:	Page:	Rev:
Rad Waste - Waste Gas	P8182L-001C	V.B.6.	14	2

Facility Learning Objective:	P8182L-001C #3
Question Source:	Direct from facility initial exam bank P8182L-001C 004
Comments:	

Record #	71
RO #	59
SRO #	57

Stem:

Given the following conditions:

- Unit 2 is at 50% power
- A load decrease is in progress per 2C1.4 "Unit 2 Power Operation"
- A lift coil fuse blows for a Control Bank "D", Group 1 rod

Which of the following describes the response of the rod control system to the next "outward" control rod demand signal?

Answers/Distracters:

a. The affected rod will DROP while the rest of Control bank "D" will MOVE OUT.

b. The affected rod will MOVE IN while the rest of Control bank "D" will MOVE OUT.

c. The affected rod will NOT MOVE while the rest of Control bank "D" will MOVE OUT.

d. ALL of Control Bank "D" rods will NOT MOVE AND an "urgent failure" alarm will come in.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-PEO	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect – As described in c below, the stationary and moveable gripper coils will energize preventing the rod from dropping.
b	Incorrect-This selection might be considered if an understanding of the sequence of coil energizing is not fully understood.
c	Correct-When rod motion is attempted, the affected rod the stationary gripper coil will energize to full current, the movable gripper coil will energize to grip the rod, the stationary gripper will then deenergize releasing the rod and the lift coil will fail to energize, resulting in no rod motion. The stationary gripper coil will then energize to grip the rod and the movable gripper will deenergize to release the rod. The rest of the rods in bank "D" will move as designed.
d	Incorrect – This selection might be considered if the blown fuse is mistakenly thought to cause an urgent failure.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
011 Control Rod Drive System	K1.03	3.4	3.6

K/A Statement:	Knowledge of the physical connections and / or cause-effect relationships between the CRDS and the following systems: - CRDM

Reference Title:	Reference #:	Section:	Page:	Rev:
Rod Control & Rod Position Indication	P8184L-005	III.C.1, VII.C	16-17, 40-45	2.5

Facility Learning Objective:	P8184L-005 #3, 5
Question Source:	Facility Initial Bank question P8184L-005 073
Comments:	

Record #	72
RO #	60
SRO #	

Stem:

(see reference)

Using the attached diagram of the Dual Channel Drawer for 1R7, which lights are lit when the Operation Selector Switch is placed in the CHECK SOURCE position?

Answers/Distracters:

a. Orange, Blue, AND BOTH Red lights

b. Orange AND Blue lights ONLY

c. Orange AND BOTH Red lights ONLY

d. Orange, Blue, AND High Alarm Red lights ONLY

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
d	1-I	2	3	1	1	Prairie Island	9/10/01

Basis for answers:

a	Incorrect-This selection could be chosen if the check source position is mistakenly thought to test all of the lights. The Low alarm light comes in on a downscale failure only.
b	Incorrect-This selection might be chosen if the source is thought to not be strong enough to deflect the meter to the high alarm position. The source strength and position is specifically chosen to give enough deflection to actuate the high alarm.
c	Incorrect-As in (d), the blue light is lit whenever the Operation Selector Switch is in any position but Operate.
d	Correct-Whenever the Operation Selector Switch is in any position but Operate the blue Channel Test light is on. In the Check Source position the check source solenoid is energized. The check source is then moved into alignment with the detector causing the high alarm to actuate. The low alarm comes in only when the meter deflects to minimum indicating a downscale failure.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
072 Area Radiation Monitoring System	A4.02	2.5	2.5

K/A Statement:	Ability to manually operate and/or monitor in the control room: - Major components

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Radiation Monitoring System	P8182L-002	VI.A	29-31	4
Radiation Monitoring System	B11	3.1B	6	5

Facility Learning Objective:	P8182L-002 #2,4
Question Source:	new
Comments:	

Record #	73
RO #	61
SRO #	

Stem:

Given the following conditions:

- Unit 1 is at 100% power
- 11 Component Cooling (CC) Pump is in service
- Unit 2 is in INTERMEDIATE SHUTDOWN with a cooldown in progress
- 21 and 22 Component Cooling (CC) pumps are in service

What occurs related to the Component Cooling Water Heat Exchangers cooling if a SI actuates on Unit 1?

Answers/Distracters:

a.
- **MV-32146**, Cooling Water Inlet to 12 CC Heat Exchanger, OPENS directly on the SI signal
- Cooling Water flow to 11, 21 and 22 CC Heat Exchangers REMAINS THE SAME

b.
- **MV-32146,** Cooling Water Inlet to 12 CC Heat Exchanger, OPENS directly when 12 CC Pump starts
- Cooling Water flow to 11, 21 and 22 CC Heat Exchangers REMAINS THE SAME

c.
- **MV-32146,** Cooling Water Inlet to 12 CC Heat Exchanger, OPENS directly on the SI signal
- Cooling Water flow to 11 CC Heat Exchanger REMAINS THE SAME
- **MV-32160** and **MV-32161**, Cooling Water Inlet to 21 and 22 CC Heat Exchangers, respectively, CLOSE and must be manually reopened to reestablish flow

d.
- **MV-32146,** Cooling Water Inlet to 12 CC Heat Exchanger, OPENS directly when 12 CC Pump starts
- Cooling Water flow to 11 CC Heat Exchanger REMAINS THE SAME
- **MV-32160** and **MV-32161**, Cooling Water Inlet to 21 and 22 CC Heat Exchangers, respectively, CLOSE and must be manually reopened to reestablish flow

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	2-RI	2	3	3	3	Prairie Island	9/10/2001

Basis for answers:	
a	Incorrect- The SI signal does not open the valve but does start the 12 CC Pump, which causes the valve to open.
b	Correct- On an SI signal, an auto start signal is sent to all ECCS and support equipment for the affected unit. This includes starting the standby CC Pump on the affected Unit. Starting a CC Pump sends a signal to open the Cooling Water Inlet Valve associated with the related CC Heat Exchanger. In this case, for Unit 1 MV-32145, Cooling Water Inlet to 11 CC HX, is open with the 11 CC Pump running, and MV-32146, Cooling Water Inlet to 12 CC HX, is closed (and the 12 CC Pump is in AUTO). Both MV-32160 and MV-32161, Cooling Water Inlet to 21 and 22 CC Heat Exchangers, respectively, are open since both CC trains are in service for the cooldown. Following the SI the only expected change in Cooling Water is isolation of the A and B headers, alignment of 121 Cooling Water Pump to the A header (Unit 1), and opening of the isolation valve to the 12 CC Hx due to starting of the 12 CC Pump.
c	Incorrect- As in (a) above, the SI signal does not open the valve but does start the 12 CC Pump, which causes the valve to open. Also the operator may think the valves close on Unit 2 since the 121 CL MVs (MV-32025, MV-32036) close to align the Pump to Loop A (also separating the headers at the Pumps discharge. Also the Aux Building cross-ties (MV-32144 & MV-32159) close to separate the A loop and B loop.
d	Incorrect-As in c above, MV-32025 and MV-32036 remain open.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
076	A3.02	3.7	3.7

K/A Statement:	Ability to monitor automatic operations of the SWS including: -Emergency heat loads

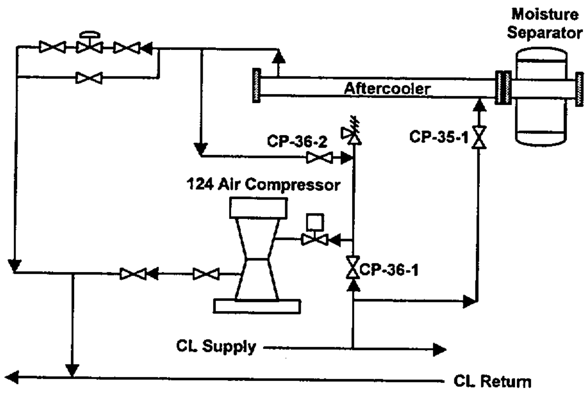
Reference Title:	Reference #:	Section:	Page:	Rev:
Cooling Water System	P8176L-003	VII.A.3.b	52-53	3
Component Cooling	P8172L-002	IV.A.1.g	11-12	4
Interlock Logic Diagram Component Cooling System		NF-40321-1		L

Facility Learning Objective:	P8176L-003 #7,9 P8172L-002 #5.a
Question Source:	New
Comments:	

Record #	74
RO #	62
SRO #	

Stem:

Given the following drawing of 124 Station Air Compressor:



What is the required valve lineup if cooling water temperature is 48° F?

Answers/Distracters:

a. **CP-35-1** → OPEN
CP-36-1 → OPEN
CP-36-2 → CLOSED

b. **CP-35-1** → CLOSED
CP-36-1 → OPEN
CP-36-2 → CLOSED

c. **CP-35-1** → OPEN
CP-36-1 → CLOSED
CP-36-2 → CLOSED

d. **CP-35-1** → OPEN
CP-36-1 → CLOSED
CP-36-2 → OPEN

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPR	2	3	2	2	Prairie Island	9/10/01

Basis for answers:

a	Incorrect – This would be the correct lineup for summer operation (>70 deg)
b	Incorrect – This would isolate cooling water to the Aftercooler, which is not desired in winter mode of operation but may be a common misconception.
c	Incorrect – This would isolate cooling water to the compressor, which is not desired in winter mode of operation but may be a common misconception.
d	Correct – Due to the cooling water temperature, the air compressor should be lined up for winter operation. This lineup tempers the water cooling the compressor jacket by first routing the cooling water through the Aftercooler.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
079 Station Air System	2.1.23	3.9	4.0

K/A Statement:	Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Reference Title:	Reference #:	Section:	Page:	Rev:
Station Air System	C34	5.12	30	17
Prairie Island Flow Diagram – Station Air/Condensate Polishing	NF-39253-3			G
Instrument and Station Air	P8178L-005	IV.A.3.d	12	2

Facility Learning Objective:	P8178L-005 #6
Question Source:	New
Comments:	

Record #	75
RO #	63
SRO #	

Stem:

(see reference)

The following conditions exist on unit 1:

- 121 & 122 air compressors are running in preferred
- 123 air compressor is in first standby
- 124 air compressor is running in preferred
- 125 air compressor is in standby
- Air systems are in their normal valve lineups

A break in a unit 1 instrument air line occurs which causes the pressure to rapidly decrease to less than 75 psig.

Which of the following correctly describes the automatic actions that occur due to this failure?

Answers/Distracters:

a.
- 123 air compressor STARTS
- **MV-32314,** U1 Instrument air header isolation, CLOSES
- **MV-32362**, 21 Air Dryer bypass, OPENS

b.
- 125 air compressor STARTS
- **MV-32318,** Service air header isolation, OPENS
- **MV-32362,** 121 Air Dryer bypass, OPENS

c.
- 123 air compressor STARTS
- **CV-39301** and **CV-39302,** Station air receiver to instrument air supply header isolation, OPEN
- **MV-32362,** 121 Air Dryer bypass, OPENS

d.
- 125 air compressor STARTS
- **CV-31740** and **CV-31741,** Instrument air to unit 1 Containment, CLOSE
- **MV-32362,** 121 Air Dryer bypass, OPENS

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	3-SPR	2	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct-As the pressure in the instrument air header decreases the compressor in 1st standby (123) starts at 90 psig. Then as pressure in the unit 1 instrument air header continues to decrease MV-32314 closes at 80 psig to align 122 air compressor to the unaffected header. 121-air dryer bypass valve (MV-32362) will then open at 78 psig to supply unit 1 instrument air header directly with unfiltered air. These auto actions isolate unit1 from unit 2 Instr air headers.
b	Incorrect-This could be considered if the operator thinks that the service air header automatically backs up the instrument air header. The service air is typically isolated from the instrument air header and if it were aligned through MV-32318 it would isolate at 85 psig when MV-32318 closes automatically. Also the service air compressors are not affected by instrument air pressure and would not start.
c	Incorrect-This could also be considered if station and instrument air were cross-connected through MV-32318, but CVs-39301 and 39302 isolate 124 &125 air compressors from the station air header. CV-39301 closes at 82 psig and CV-39302 will not open until service air pressure recovers to greater than 88 psig.
d	Incorrect- As in b above, the service air compressor (125) is not affected by the decrease in instrument air pressure. Also the containment isolation valves CVs-31740 and 31741 are not affected either. They close when Containment pressure reaches 17 psig.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
079	Station Air System (SAS)	K4.01	2.9	3.2

K/A Statement:	Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: -Cross-connect with IAS

Reference Title:	Reference #:	Section:	Page:	Rev:
Instrument and Station Air	P8178L-005	V.C	21	2
STATION AIR SYSTEM	C34	1.0	3	17
LOSS OF INSTRUMENT AIR	C34 AOP1	2.2.	3	10

Facility Learning Objective:	P8178L-005 #2,7
Question Source:	New
Comments:	Figure B34-01 needed as reference

Record #	76
RO #	64
SRO #	

Stem:

Given the following conditions:

- ONE thermal detector associated with Fire Protection Zone 12 (Relay Room) has failed
- This failure has resulted in an ALARM condition for Zone 12

As a result of this failure, when is carbon dioxide released to the Relay Room?

Answers/Distracters:

a. Following a 60-second time delay from a SECOND detector going into an alarm condition.

b. Immediately IF a SECOND detector goes into an alarm condition.

c. Following a 60-second time delay from the alarm actuation.

d. Immediately upon the alarm actuation.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-F	2	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- A second detector actuation is NOT required to actuate the system.
b	Incorrect-- A second detector actuation is NOT required to actuate the system.
c	Correct-The Cardox System provides a total flood capability for the Relay/Computer Room in the event of a fire. Various thermal detectors are located throughout the Relay/Computer Room. If one of the detectors reaches 140°F, an automatic actuate signal is sent to the system. Upon receipt of either an automatic (or a manual actuate signal), an alarm and a 60 second timer are actuated. The timer provides a delay period prior to system actuation to ensure all personnel have adequate time to evacuate the room. Following the 60 second time delay, the system dumps CO2 to the room for 3-1/2 minutes, goes into a soak mode for 1-1/2 minutes, and then shuts down
d	*Incorrect-* The CO2 release does not occur immediately upon the alarm. The alarm indicates a 60-second timer has been initiated, which initiates the release at the end of 60 seconds.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
086	Fire Protection System (FPS)	K6.04	2.6	2.9

K/A Statement:	Knowledge of the effect of a loss or malfunction of the following will have on the Fire Protection System: -Fire, smoke, and heat detectors

Reference Title:	Reference #:	Section:	Page:	Rev:
Fire Detection and Protection Systems	P8178L-002	III.O.6 & 7	19	2
FIRE PROTECTION SYSTEM	B31A	3.18.B	30-31	3

Facility Learning Objective:	P8178L-002 #5,7
Question Source:	New
Comments:	

Record #	77
RO #	
SRO #	58

Stem:

(see reference)

Given the following conditions on Unit 1:

- The Unit was at 92% power during a power ascension to 100% power
- Control Rod Bank D began stepping out at maximum speed
- The operators have initiated actions of 1C5 AOP1"UNCONTROLLED WITHDRAWAL OF AN RCCA"
- Control rod motion stopped when the Rod Bank Selector was taken to MAN
- Pressurizer pressure is 2330 psig
- Tavg is 570°F
- ΔT is 70°F
- ΔI is + 6.2% of target ΔI
- QPTR is 1.02

What action is required?

Answers/Distracters:

a. RCS pressure MUST be **reduced** to LESS THAN 2250 psig within 5 minutes.

b. ΔI MUST be **reduced** to +/- 5% of target ΔI within 15 minutes.

c. Tavg MUST be **decreased** below 565°F AND the Unit placed in MODE 3 within ONE hour.

d. Reactor power MUST be **reduced** to AND maintained below 50% within 2 hours.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	3-PEO	1	3	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is the action for exceeding pressure Safety Limit of 2735 psig if the Unit is in MODE 3, 4 or 5. RCS pressure is abnormally high but well below the Safety Limit.
b	Correct- Delta-I is required to be maintained within +/- 5% of the target ΔI when reactor power is above 90%. When indicated AFD deviates from the target band, then it must be restored within the band within 15 minutes or THERMAL POWER reduced to less than 90%. The Target Band is +/- 10% below 90% power.
c	Incorrect- Tavg is also part of the factors used to determine the Reactor Core Safety Limit, along with Przr pressure based on the Figure. The point determined by the given values is below the 2235 psig pressure curve and within the "operating" limits.
d	Incorrect-- This is action required if QPTR exceeds 1.07 combined with the time limit associated with QPTR exceeding 1.02. Operation with QPTR at 1.02 is acceptable.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
001	Continuous Rod Withdrawal	2.1.11	3.0	3.8

K/A Statement:	Conduct Of Operations -Knowledge of less than one hour technical specification action statements for systems.

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	VI.A.3.b.11	49	4
Prairie Island Technical Specifications		3.10.B.4 & 5	TS.3. 10-3	136
Core Operating Limits Report Unit 1, Cycle 21		Axial Flux Difference Limits	2	1

Facility Learning Objective:	P8184L-002 #18,19
Question Source:	New
Comments:	Figure TS.2.1-1, COLR Figure 3 required

Record #	**78**
RO #	**65**
SRO #	**59**

Stem:

Given the following conditions on Unit 1:

- The Unit is at 88% power <u>AND</u> holding for a calorimetric during a power ascension to 100% power
- Control rods were in auto when a single RCCA in Bank D began stepping out at maximum speed
- The operators have initiated actions of 1C5 AOP1"UNCONTROLLED WITHDRAWAL OF AN RCCA"
- Control rod motion stopped when the Rod Bank Selector was taken to MAN
- The following readings were taken from the Power Range NIS cabinets:

	N41	N42	N43	N44	
Det. A (upper)	375.0	360.0	365.0	360.0	(microamperes)
Det. B (lower)	350.0	345.0	370.0	340.0	(microamperes)

- A full power current on all detectors is known to be 400.0 microamperes.

Which detector has the highest quadrant power tilt ratio (QPTR)?

Answers/Distracters:

a. N41 upper

b. N42 upper

c. N43 lower

d. N44 lower

Answer:	***LOK***	***Tier:***	***LOD***	***RO Group:***	***SRO Group:***	***Facility:***	***Exam Date:***
c.	3-SPK	1	4	2	1	Prairie Island	9/10/2001

Basis for answers:

a, b, &d	Incorrect- A) B) & D represent other selectable answers depending on the understanding of QPTR. N41 may be selected since it is the actual highest value for current.
c	Correct- QPTR is defined as the maximum excore detector output divided by the average of all four detector outputs. QPTR = { I(high)/I(100%)] / {[I(41) + I(42) + I(43) + I(44) / I(100%)] / 4}. This is performed for both the top and bottom detectors. For the above case: N41Upper - QPTR= (375.0/400.0) / {[(375.0+360.0+365.0+360.0) / 400] /4} = {.9375 / [(1460/400)/4]} = .9375 / .9125 = 1.027. N42 Upper - (360.0/400.0) / {[(375.0+360.0+365.0+360.0) / 400] /4} = .9000/.9125 = 0.9863. N43 Lower - QPTR= (370.0/400.0) / {[(370.0+345.0+370.0+340.0) / 400] /4} = {.925 / [(1425/400)/4]} = .9375 / .8906 = 1.053. N44 Lower - QPTR =(340.0/400.0) / [(370.0+345.0+370.0+340.0) / 400] /4} = .8500 / .8906 = 0.9544. The highest QPTR value is thus associated with N43 lower detector.

K/A System/Evolution:		***K/A #:***	***KAVRO***	***KAVSRO***
001	Continuous Rod Withdrawal	AK1.11	2.8	3.3

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: -Definitions of core quadrant power tilt

Reference Title:	***Reference #:***	***Section:***	***Page:***	***Rev:***
Reactor Heat & Thermal Limits	P8159L-058	V.C	19-20	2
Nuclear Instrumentation System	P8184L-002	IV.D.1	35-37	4
Prairie Island Technical Specifications		1.0	TS.1-5	122

Facility Learning Objective:	P8159L-058 #15 P8184L-002 #14
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8184L-002 020. Layout of premise and question was changed to "standard" layout.

Record #	79A
RO #	
SRO #	60

Stem:

Given the following conditions on Unit 1:

- An ATWS has occurred AND E-0 "Reactor Trip OR Safety Injection" was entered
- The operators then initiated the actions of 1FR-S.1 " Response To Nuclear Power Generation / ATWS "
- NO Adverse Containment parameters exist

Which of the following conditions identify the parameter(s) that must be satisfied in order to transition back to 1E-0 "Reactor Trip OR Safety Injection"?

Answers/Distracters:

a. The Cold Shutdown Boron Concentration value is achieved.

b. ALL Control AND Shutdown Bank rods indicate fully inserted.

c. BOTH reactor trip breakers AND BOTH trip bypass breakers are verified open.

d. Power Range NIS channels indicate less than 5% AND Intermediate Range SUR is negative.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-P	1	2	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Boration is initiated and a CAUTION exist in FR-S.1 prior to transition that directs boration continue to obtain adequate shutdown margin during subsequent actions; however, the Cold Shutdown value is NOT required to be satisfied for transition.
b & c	Incorrect- Attempts are made in performance of the procedure to insert rods and to open the reactor trip breakers; however, these are NOT satisfactory measures for ensuring the reactor is shutdown and are not required to be met to transition.
d	Correct- The parameters for verifying subcriticality are those indicated on the PR NIS and a negative SUR on IR NIS

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
029	Anticipated Transient Without Scram (ATWS)	2.4.8	3.0	3.7

K/A Statement:	Emergency Procedures / Plan -Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.

Reference Title:	Reference #:	Section:	Page:	Rev:
F/FR Review	P8197L-014	II.B.6.d	15	2
Response To Nuclear Power Generation / ATWS	1FR-S.1	13 & 14	7	11

Facility Learning Objective:	P8197L-014 #1,3
Question Source:	NRC Exam Bank
Comments:	Salem 2/1999 NRC EXAM SRO question 73

Record #	79B
RO #	66
SRO #	

Stem:

(see reference)

Given the following conditions on Unit 1:

- A SG tube rupture has been diagnosed on 12 SG
- Buses 11 and 12 have failed to transfer to 1R transformer following the trip
- 1E3 "Steam Generator Tube Rupture" has been completed up to the steps initiating RCS depressurization (Steps 19 and 20 provided)
- Both Pressurizer PORVs will NOT open
- The charging line to Pressurizer ΔT is 500°F

What is the correct action to be taken based on the above information?

Answers/Distracters:

a. **Start** the 11 RCP in order to establish normal Pressurizer spray.

b. **Establish** normal letdown to reduce the spray line ΔT to less than 320°F.

c. **Establish** auxiliary spray flow until the condition is met to stop depressurization.

d. **Perform** the actions of 1ECA-3.3 "SGTR Without Pressurizer Pressure Control" since adequate Pressurizer pressure control CANNOT be established.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-SPR	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Neither RCP is currently available for use with the 11 and 12 bus deenergized. Note that either RCP could be started if available, not just the 11 RCP.
b	Incorrect-- This is the preferred method DURING NORMAL OPERATIONS to reduce the spray DT when using aux spray. This option is not discussed nor should it be used since it does not follow the procedural direction and presents other considerations that will slow the achievement of equilibrium between the primary and secondary systems and stop outleakage.
c	Correct-Procedural (1C1-3, Limitiation4.4) and Technical Specification (TS 3.2.b) provides for a limitation of DT at a maximum of 320°F between the Pressurizer and the spray fluid if spray is to be used. However, in an accident condition the crew is to realize EOP procedural directions are to be followed and that Tech Specs may be exceeded to place the plant is a safer condition. Since depressurization is mandated to minimize (and stop) the leakage from primary to secondary, the limitation may be exceeded, provided other methods of depressurization are not available
d	Incorrect- This progression should only be used if NO method of pressure control can be established. This may be considered if the operator determines that the spray DT limitation is applicable.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
038	Steam Generator Tube Rupture (SGTR)	2.4.13	3.3	3.9

K/A Statement:	Emergency Procedures / Plan -Knowledge of crew roles and responsibilities during EOP flowchart use.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-3 Series Review	P8197L-13A	VI.C.1,X.D.3.f	41-42, 76-77	2
Steam Generator Tube Rupture	1E-3	Steps 19 & 20	12-13	15
Background Information for Steam Generator Tube Rupture	1E-3	1st Note Step 20	11	15

Facility Learning Objective:	P8197L-13A #12 & 15
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8197L-013 009. Changed premise format. Simplified two selections to include temperature differential value.

Record #	80
RO #	67
SRO #	61

Stem:

Given the following conditions on Unit 1:

- The Unit is at 80% power
- The operators have entered 1C5 AOP5 "MISALIGNED ROD, STUCK ROD, AND/OR RPI FAILURE OR DRIFT"
- The operator is reviewing the symptoms that indicate a possibly misaligned RCCA.

Which indication would NOT be present if the problem is a stuck RCCA?

Answers/Distracters:

a. RPI AND Group Step Counter in disagreement.

b. Abnormal flux tilt indicated on Power Range NIS.

c. Movement shown on the suspect rod RPI as the IN-HOLD-OUT switch is operated.

d. Movement shown on the suspect rod Group Step Counter as the IN-HOLD-OUT switch is operated.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-F	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The RPI & Group Step Counter are expected to be different for the stuck rod and a misaligned rod, as is indicative of the alarm condition that would require AOP entry.
b	Incorrect-- Abnormal flux tilt may be expected for a RCCA that is misaligned or stuck rod, as is indicative of the alarm condition that would require AOP entry
c	Correct- Three of the listed symptoms are common to either a stuck or misaligned rod. However if the rod is stuck, no movement is indicated on the suspect rod position indicator.
d	Incorrect- The Group Step Counter has input from the Logic Cabinet that counts demanded movement of the Group. Motion of the counter is expected whenever rod motion is demanded in either the stuck rod or misaligned rod condition.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
005	Inoperable/Stuck Control Rod	AA1.01	3.6	3.4

K/A Statement:	Ability to operate and/or monitor the following as they apply to the inoperable/Stuck Control Rod: -CRDS

Reference Title:	Reference #:	Section:	Page:	Rev:
Rod Control & Rod Position Indication	P8184L-005	IX.B	49-50	2
MISALIGNED ROD, STUCK ROD, AND/OR RPI FAILURE OR DRIFT	1C5 AOP5	2.1.3 2.1.4	2-4	3

Facility Learning Objective:	P8184L-005 #3,6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8184L-005 021. Layout of premise and question was changed to "standard" layout.

Record #	81
RO #	68
SRO #	

Stem:

Given the following conditions on Unit 1:

- A Pressurizer Safety Valve has FAILED OPEN
- SI has actuated
- The actions of 1E-0 and 1E-1 are complete
- The actions of 1ES-1.1 "Post-LOCA Cooldown And Depressurization" are being performed
- RCS Subcooling is 0°F
- Pressurizer level indicates 100%
- RCPs have been stopped
- RVLIS Full Range indicates 38%
- Containment pressure is 7 psig
- SG levels are 6% and 10% narrow range
- SI Pumps are running

What is the proper set of actions for the given conditions?

Answers/Distracters:

a. **Continue** in 1ES-1.1 AND **restart** one RCP.

b. **Enter** ES-0.2 "SI Termination" in order to **stop** the SI Pumps.

c. **Enter** FR-C.2 "Response To Degraded Core Cooling" to **establish** adequate core cooling.

d. **Enter** 1ES-0.3A "Natural Circulation Cooldown With CRDM Fans" to **verify** natural circulation flow.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-DR	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- If degraded core cooling conditions did not exist, the next section of ES-1.1 directs RCS depressurization (to restore Przr level) and attempt to restart a RCP
b	Incorrect-- The Information Page of ES-1.1 provides conditions for evaluation of transition to ES-0.2. These conditions are not met for subcooling (> 35°F) or RCS pressure (>2000 psig OR stable or increasing).
c	Correct- The conditions give indicate a loss/degradation of core cooling. The path followed is F-0.2 Core Exit TC's < 1200°F; RCS subcooling < 20°F; No RCPs are running; Core Exit TC's < 700°F; and, RVLIS Full Range < 40%. This directs the operator to FR-C.2.
d	Incorrect-- ES-0.3A is entered from a non-accident condition when the RCPs are not running (and not available). This is not an appropriate action for an accident condition - LOCA

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)	2.4.8	3.0	3.7

K/A Statement:	Emergency Procedures / Plan -Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.

Reference Title:	Reference #:	Section:	Page:	Rev:
EOP Intro-Procedure Review	P8197L-010	III.E.2	19-20	2
E-1/E-2 Review	P8197L-012	VI.B.2,VII.C	41,49	2
Core Cooling (Status Tree)	F-0.2			4

Facility Learning Objective:	P8197L-010 #4 P8197L-012 #26
Question Source:	New
Comments:	

Record #	82
RO #	69
SRO #	62

Stem:

Given the following conditions on Unit 1:

- The Unit is at 100% power
- Component Cooling (CC) to 11 SI Pump has been isolated to stop a CC system leak
- A LOCA then occurred on Unit 1 reducing RCS pressure to 1400 psig
- All Safeguards Equipment responded as expected to the SI actuation signal

What is the preferred action regarding 11 SI Pump?

Answers/Distracters:

a. **Allow** 11 SI pump to run as long as the Unit Coolers for the SI Pumps are operating.

b. **Allow** 11 SI pump to run for up to three hours without CC flow.

c. **Stop** 11 SI pump since the pump can NOT be operated without CC flow.

d. **Stop** 11 SI pump since 12 SI Pump is operating properly.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-F	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- There is no tie to operation of the Pump with/without the Unit Coolers although the Cooler is expected to run during the accident, providing cooling to the area. Containment Spray pumps may be operated indefinitely w/o CC flow.
b	Incorrect-- No time limit is specified for running the pump, but operation, if required, is based on oil temperatures. The RHR Pumps have a 3 hour operating time limit with RHR (RCS) temperatures at 300°F.
c	Incorrect- The pump may be operated without CC flow, if needed (i.e., the other SI Pump not operating and SI injection flow required), provided the oil temperatures remain within operating limits.
d	Correct- Table 1 of 1C14 AOP1 provides direction for determining the CC requirements for the specific plant condition. The CAUTION prior to the table states, "ONE (1) TRAIN OF SAFEGUARDS EQUIPMENT IS REQUIRED TO MITIGATE ACCIDENT CONDITIONS. IF CC FLOW IS LOST TO ONE COMPONENT BUT IS PROVIDED TO THE ALTERNATE TRAIN COMPONENT, THEN IT IS PREFERABLE TO ONLY OPERATE THE COMPONENT WITH CC FLOW DURING AN ACCIDENT." Otherwise, during an accident condition, the operating limits are, "Maintain bearing oil supply temperatures (local) between 120°F and 180°F."

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
009	Small Break LOCA	EA1.05	3.4	3.4

K/A Statement:	Ability to operate and/or monitor the following as they apply to a small break LOCA: -CCWS

Reference Title:	Reference #:	Section:	Page:	Rev:
Component Cooling	P8172L-002	IV.4.I, V.C.1	17, 28-29	4
LOSS OF COMPONENT COOLING	1C14 AOP1	Table 1	8	8

Facility Learning Objective:	P8172L-002 #2.c, 6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8172L-002 036. Changed premise layout and question.

Record #	**83**
RO #	**70**
SRO #	**63**

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- SI has actuated
- The actions of 1E-0 "Reactor Trip Or Safety Injection" are being performed
- Both SG Narrow Range levels are offscale low
- 11 SG pressure is 835 psig AND decreasing slowly
- 12 SG pressure is 885 psig AND decreasing slowly
- Containment pressure is 4 psig
- 11 RCS cold leg temperature is 521°F AND decreasing slowly
- 12 RCS cold leg temperature is 530°F AND decreasing slowly

Which of the following is correct about the SG pressures?

Answers/Distracters:

a. NO SG pressure is decreasing in an uncontrolled manner.

b. ONLY 11 SG pressure is decreasing in an uncontrolled manner.

c. ONLY 12 SG pressure is decreasing in an uncontrolled manner.

d. BOTH SG pressures are decreasing in an uncontrolled manner.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
a	3-SPR	1	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct-Both 11 & 12 SGs have pressures, which are higher than saturation pressures for the RCS cold leg temperatures given, therefore, the RCS, is cooling the SGs causing the pressures to decrease. It can therefore be determined tha the SGs are not depressurizing uncontrollably. The basis for step 19 of 1E-0 is to determine if a faulted SG exists. An incorrect response would be to have the operator transition to 1E-2"Faulted Steam Generator Isolation"
b *c* *d*	See above

K/A System/Evolution:		*K/A #:*	*KAVRO*	*KAVSRO*
009	Small Break LOCA	EK1.02	3.5	4.2

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: -Use of steam tables

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
E-0 Review	P8197L-011	II.B	9-10	2
Reactor Trip Or Safety Injection	1E-0	Step 19	13	19
Steam Tables				

Facility Learning Objective:	P8197L-011 #6
Question Source:	New
Comments:	Steam Tables needed for exam

Record #	84
RO #	
SRO #	64

Stem:

Given the following conditions on Unit 1:

- RCS Pressure is 360 psig
- RCS temperature is 118°F
- 11 and 12 SG skin temperatures are 75°F
- The crew is in the process of completing 1D8 "Unit 1 Filling And Venting The Reactor Coolant System".
- 12 RCP has been standing idle for 30 minutes after 2 unsuccessful attempts to complete the required 5-minute run due to RCS pressure drop.
- The RCS system engineer recommends an immediate third start to complete the 5-minute run and sweep air from the SG U-Tubes

Based on this information, what action should the operators take concerning 12 RCP?

Answers/Distracters:

a. Immediately **start** 12 RCP since the RCS is within 50°F of 12 SG skin temperature.

b. Do NOT start 12 RCP until the RCS is within 40°F of 12 SG skin temperature to minimize the RCS pressure transient.

c. Do NOT start 12 RCP until at least 45 minutes has elapsed since the last start to prevent the motor windings from overheating.

d. Immediately **start** 12 RCP since greater than 20 minutes has elapsed since the last start.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-P	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- One of the limitation for the Reactor Coolant Pumps is "To minimize pressure transients, the RCS and Steam Generators must be within 50°F of each other before starting a RCP". Although the RCS and SGs are within this 50°F limit the RCP does not meet the required time limits for starts on a RCP as described in c below.
b	Incorrect- The RCS to SG skin temperature limit is 50°F as described in a above. Since the RCS to SG temps are within 50°F of each other this limitation is met and therefore is not a concern for starting the RCP at this time.
c	Correct-One of the limitations for starting a RCP is "To prevent damage to the motor windings, only two starts are permitted at ambient temperature. To limit starting current, the motor must have coasted to a full stop. A third start may be attempted if the motor has been running 20 minutes or standing idle for 45 minutes after the second start. This allows the motor windings time to cool down-the motor cools more efficiently when running." Since the RCP has been standing idle for only 30 minutes it is not appropriate to start the RCP.
d	Incorrect-As described above the 20-minute time concern is a running time. Since the RCP ran for less than 5 minutes this condition is not met. It is not appropriate to start the RCP at this time.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
015 Reactor Coolant Pump (RCP) Malfunctions	2.1.32	3.4	3.8

K/A Statement:	Conduct Of Operations:-Ability to explain and apply all system limits and precautions

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Coolant Pumps	P8170L-002	VI.D	27	2
Unit 1 Filling and Venting the RCS	1D8	4.4.5	6	15
Prairie Island Technical Specifications		T.S.3.1.A.1.c(4)	T.S.-3.1.2	135
Reactor Coolant Pumps	1C3	4.0	5	26

Facility Learning Objective:	P8170L-002 #5
Question Source:	New
Comments:	

Record #	85
RO #	71
SRO #	65

Stem:

Given the following conditions on Unit 1:

- The Unit is at 15% power
- The following annunciators are in alarm:
    - **47015-0206** 11 RCP LABYRINTH SEAL LO DP
    - **47015-0207** 12 RCP LABYRINTH SEAL LO DP
    - **47015-0208** 11 RCP NO. 1 SEAL INLT OR OUTL HI TEMP
    - **47015-0209** 12 RCP NO. 1 SEAL INLT OR OUTL HI TEMP
    - **47015-0409** SEAL WATER INJECTION FILTER HI DP
- Seal injection flows to each RCP indicate LESS THAN 1 gpm
- CV-31245, 11 RC Pump Thermal Barrier Clnt Outl valve has failed CLOSED
- RCP lower bearing water temperatures indicate:
    - 211°F AND increasing for 11 RCP
    - 181°F AND increasing for 12 RCP

Per C3 AOP2, "Loss of Reactor Coolant Pump Seal Cooling", which of the following actions are required at this time?

Answers/Distracters:

a. **Trip** the reactor AND **stop** both RCPs.

b. **Trip** the reactor, **stop** 11 RCP only AND **monitor** 12 RCP bearing water temperature.

c. **Shutdown** the reactor within ONE hour AND then **stop** both RCPs.

d. **Stop** 11 RCP only, **shutdown** the reactor within ONE hour AND **monitor** 12 RCP bearing water temperature.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	2-RI	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- There is no current condition that requires stopping the 12 RCP.
b	Correct- Both RCPs have experienced failures. Seal Injection flow is lost to both RCPs. Additionally, CC flow to the 11 RCP Thermal Barrier has been lost. At 200°F bearing water temperature, the operators are directed to trip the reactor (enter E-0) and then stop the affected RCP. As directed in earlier action with bearing temperatures < 200°F for an affected RCP, actions are continued to restore CC and/or seal injection flow and the bearing water temperature is monitored. It is expected with "normal" CC flow to the thermal barrier, the RCP may continue to be operated.
c	Incorrect- Temperature greater than 200 deg on a RCP bearing requires tripping the reactor and stopping of the affected RCP. It is not acceptable to delay long enough to do a shutdown.
d	Incorrect-- Reactor operation is not allowed with one RCP secured. The ONE hour TS requirement is allowed for shutdown of cooling loops with the reactor subcritical and RCS temperature < 350°F.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
015 Reactor Coolant Pump (RCP) Malfunctions	AK3.03	3.7	4.0

K/A Statement:	Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions: - Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction

Reference Title:	Reference #:	Section:	Page:	Rev:
Reactor Coolant Pumps	P8170L-002	VI.D	27	2
LOSS OF RCP SEAL COOLING	1C3 AOP2	2.4.2	4	4

Facility Learning Objective:	P8170L-002 #4, 8
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Initial Bank question P8170L-002 040Changes to premise include loss of CC flow affecting only one RCP and temperature values for seal water (bearing) temperatures given. 3 selections, correct answer changed - (only one RCP required to be stopped).

Record #	**86**
RO #	**72**
SRO #	

Stem:

(see reference)

Given the following conditions on Unit 1:

- A LOCA has occurred from 100 % power
- The crew has completed the actions of 1E-0 "Reactor Trip or Safety Injection" AND has entered 1E-1 "Loss of Reactor or Secondary Coolant"
- The following parameters are noted
  - Power Range NIS indicate 1% power
  - Intermediate Range SUR indicate + 0.1 dpm
  - RCS pressure is 1200 psig
  - RCS cold leg temperature 200°F
  - SG levels: 40% WR (11) and 48% WR (12)
  - Feed flow: 0 gpm (11) and 50 gpm (12)

Which of the following statements describes the proper procedure flow path the operator should take?

Answers/Distracters:

a. **Remain** in 1E-1 "Loss of Reactor or Secondary Coolant".

b. **Transition** immediately to 1FR-S.1 "Response to Nuclear Generation/ATWS".

c. **Transition** immediately to 1FR-H.1 "Response to Loss of Secondary Heat Sink".

d. **Transition** immediately to 1FR-P.1"Response to Imminent Pressurized Thermal Shock Condition".

Answer:	***LOK***	***Tier:***	***LOD***	***RO Group:***	***SRO Group:***	***Facility:***	***Exam Date:***
c	3-SPR	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This would be a valid option if the operator determined NO RED or ORANGE path existed
b	Incorrect- This would be valid if a RED path was determined for Subcriticality or if the Subcriticality is erroneously determined to be the highest AFFECTED CSF.
c	Correct- The operator will use the given information and evaluate the provided Critical Safety Function Status Trees to determine the end path condition. These conditions provide 1) ORANGE path for Subcriticality; 2) RED path for Heat Sink, 3) RED path for Integrity. The CSFSTs are also applicable once transition occurs from E-0. Once the operator determines this, the CSFTS are evaluated and the pathways determined. The operator should transition to the procedure that addresses the Heat Sink CSF (1FR-H.1).
d	Incorrect-- This is considered since a RED path does exist and if the Heat Sink is incorrectly prioritized.

K/A System/Evolution:		***K/A #:***	***KAVRO***	***KAVSRO***
E07	Saturated Core Cooling	2.4.21	3.7	4.3

K/A Statement:	Emergency Procedures / Plan -Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.

Reference Title:	***Reference #:***	***Section:***	***Page:***	***Rev:***
EOP Intro-Procedure Review	P8197L-010	III.E.2	19-20	2
Heat Sink CSF	F-0.3		1	3

Facility Learning Objective:	P8197L-010 #4,5
Question Source:	Facility Exam Bank
Comments:	**Material Required for Examination** F-0.1, F-0.3 and F-0.4 Status Trees only. Initial Bank question P8197L-014 003. Modified from given the Path end condition (RED, ORANGE, etc.) to give values for those parameters used to evaluate and determine the proper path.

Record #	**87**
RO #	**73**
SRO #	**66**

Stem:

Which of the following conditions requires entry into C12.5 AOP1, "EMERGENCY BORATION OF THE REACTOR COOLANT SYSTEM"?

Answers/Distracters:

a.
- The Unit is at 50% power
- Instrument Air is lost to the CVCS makeup valves
- Annunciator **47013-0207** CONTROL BANKS LO-LO LIMIT is in alarm

b.
- The Unit is at 20% power
- A loss power to Bus 15 has occurred
- Annunciator **47013-0107** BANK D ROD WITHDRAWAL HI LIMIT is in alarm

c.
- The Unit is in MODE 3 at normal operating temperature and pressure
- Core exposure is 12 GWD/MTU
- A loss of Train A DC power has occurred
- RCS boron concentration is reported at 1220 ppm

d.
- The Unit is in MODE 5
- Core exposure is 12 GWD/MTU
- Two charging pumps are operating
- Instrument Air is lost to the CVCS charging pumps
- RCS boron concentration is reported at 1800 ppm

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a.	3-SPK	1	4	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- With the control bank below the rod insertion limit as indicated by the alarm for 0207, boration is required to restore rod position above the RIL. Normal boration would normally be used; however, with a failure of IA the normal boration flowpath is not available. In the normal boration line, CV-31155 fails open but the series valve CV-31200 fails closed preventing boric acid flow to the charging pumps suction.
b	Incorrect- The alarm condition requires operation of the Make-up system with dilution specified to allow insertion of Control Bank D. Also the loss of Bus 15 does not affect operation of the Makeup system since it and its valves are powered from the DC system. A loss of the Bus 15 power would affect the Emergency Boration valve MV-32086 and the 11 BAT Pump.
c	Incorrect- A loss of DC power would affect the operation of the Makeup system including boration; however, the given condition does not require boration per Figure C1-10A.
d	Incorrect- Loss of air to the charging pump affects boration capability in that the charging pumps will fail to minimum speed on loss of air. However, the given condition does not require boration per Figure C1-10A, although it would per the chart if 3 pumps were operating.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
024	Emergency Boration	2.4.4	4.0	4.3

K/A Statement:	Emergency Procedures / Plan including: -Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Reference Title:	Reference #:	Section:	Page:	Rev:
CVCS	P8172L-001A	V.D	36-37	3
EMERGENCY BORATION OF THE REACTOR COOLANT SYSTEM	C12.5 AOP1	2.1	2	2
Alarm Response Procedure	47013	47013-0207	1	27

Facility Learning Objective:	P8172L-001A #9
Question Source:	New
Comments:	

Record #	**88**
RO #	
SRO #	**67**

Stem:

Given the following conditions on Unit 1:

- Unit 1 is at 100% power
- RCS Tavg is stable at 560°F
- VCT level is stable at 33%
- VCT pressure is 20 psig
- Annunciator **47020-0203** 11 CC SURGE TANK HI/LO LVL is in alarm
- 11 CC Surge Tank indicates –10"

Which of the following is a correct action based on the given conditions?

Answers/Distracters:

a. Within one hour **close CV-31245** OR **CV-31246**, the affected RCP THERM BARRIER CLNT OUTL valve AND **initiate** shutdown of Unit 1.

b. Within one hour **initiate** shutdown of Unit 1 due to exceeding the total RCS leakage limit via the letdown heat exchanger.

c. **Initiate** boration of the RCS to maintain the reactor core safety limits due to leakage in the seal water heat exchanger.

d. **Initiate** sampling of the Spent Fuel Pool to verify the required boron concentration due to leakage in the spent fuel pool heat exchanger.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
d.	2-DS	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- If leakage in thermal barrier Hx were to occur it would cause leakage out of the RCS and into the CC system causing CC surge tank level to rise.
b	Incorrect- If leakage in the letdown HX occurs, leakage is expected into the CC system until letdown is isolated.
c	Incorrect- If leakage occurs in the Seal Water HX, then outleakage from CC occurs and the potential for RCS dilution does exist. The conditions with stable VCT level indicates that leakage into the CVCS is not occurring.
d	Correct- The normal indicated CC Surge Tank level is between +2" and +10" (in the control room). The annunciator alarms at -6" decreasing level. Therefore outleakage from the CC system is expected to have occurred. The SFP has no direct connection to the RCS and therefore leakage in that heat exchanger would result in decreasing CC Surge Tank level, but would have no effect on RCS parameters (RCS temp, VCT level, Przr level).

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
026 Loss of Component Cooling Water (CCW)	AA2.02	2.9	3.6

K/A Statement:	Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: -The cause of possible CCW loss

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Component Cooling	P8172L-002	V.C.3	29	4
LOSS OF COMPONENT COOLING	1C14 AOP1	2.4.5.A	5	8

Facility Learning Objective:	P8172L-002 #2.e, 6
Question Source:	new
Comments:	

Record #	89
RO #	74
SRO #	68

Stem:

Given the following conditions on Unit 2:

- The Unit is at 100% power
- Pressurizer pressure is 2235 psig
- One backup heater group ON, variable heater group ON
- Pressurizer Pressure Control Selector switch is in the 2-1 position
- Pressurizer Pressure Control in AUTO

What would be the response of Pressurizer pressure control to a single Pressurizer Spray Valve controller FAILURE to 100% output?

Answers/Distracters:

a. Pressurizer pressure does NOT decrease because the spray valves do NOT open below 2260 psig.

b. Pressurizer pressure decreases to 2215 psig where the variable heater stabilizes pressure.

c. Pressurizer pressure decreases to 2210 psig, where all backup heaters turn on and stabilize pressure.

d. Pressurizer pressure decreases to 1900 psig where an automatic reactor trip occurs.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	1-I	1	2	1	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is normally true for the Pressurizer Pressure controller, which provides a demand signal to the Spray Valve controllers when the demand signal reaches 62.5% rising pressure (2260 psig equivalent). In this case the output signal is downstream of the Pressurizer Pressure controller, so it does not affect the valve position (demanded 100% open).
b	Incorrect- Pressure will fall resulting in the actuations described for the Przr heaters. However, the fully open spray valve provides cooling at a rate greater than the capacity of all Przr heaters.
c	Incorrect - Pressure will fall resulting in the actuations described for the Przr heaters. However, the fully open spray valve provides cooling at a rate greater than the capacity of all Przr heaters.
d	Correct- The Spray controller failure will cause the Spray Valve to go to the full open position. Przr pressure will begin to drop. When Przr pressure drops to 2215 psig, the Przr Pressure controller will output a demand signal that applies full power to the variable heaters. Pressure will continue to fall. At 2210 psig, the Przr Pressure controller will output a demand signal that energizes the backup heaters. Due to the spray, Pressurizer pressure will continue to fall until the reactor trips at 1900 psig (2/4 channels).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
027 Pressurizer Pressure Control (PZR PCS) Malfunction	AK2.03	2.6	2.8

K/A Statement:	Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Reference Title:	Reference #:	Section:	Page:	Rev:
Pressurizer Pressure Control System	P8170L-005	IV.A.2 & 3	11-14	4
Reactor Coolant System	P8170L-003	IV.A.2.a.5	20	4

Facility Learning Objective:	P8170L-005 #3, 4 P8170L-003 #2.6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8170L-005 006. Modification to premise layout and change question to include failure in stem.

Record #	90
RO #	75
SRO #	

Stem:

Given the following conditions on Unit 1:

- The Unit has tripped
- The CONTROLLING Pressurizer level channel has failed LOW
- RCS Tavg 540°F
- RCS pressure 2000 psig
- Actual Pressurizer level is 50%
- Pressurizer liquid temperature 620°F
- Pressurizer vapor temperature 634°F
- The alternate Pressurizer level control channel has been selected AND all Pressurizer heaters have been turned on

Which one of the following describes the present state of the Pressurizer?

Answers/Distracters:

a. Superheat conditions exist in the pressurizer, BUT heaters and sprays will maintain pressure.

b. The pressurizer is at equilibrium saturation conditions AND normal pressure control is available with heater and sprays.

c. The pressurizer liquid is subcooled with pressurizer heaters maintaining pressure.

d. The pressurizer liquid is subcooled AND pressure is being maintained by charging flow compressing the vapor space.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	3-SPK	1	4	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The Przr conditions are subcooled not superheated. Heaters & sprays are available but the heaters have not yet created saturation conditions.
b	Incorrect- Saturation conditions do not exist in either the liquid or vapor space in the Przr, due to overfilling.
c	Incorrect- The heaters may be raising the temperature of the incoming water, but they are not maintaining pressure as the saturation pressure associated with the liquid is 1772 psig.
d	Correct- With the vapor space at 634°F the saturation pressure is 1960 psig and the liquid saturation pressure (620°F) is 1772 psig. This indicates the water (and vapor) in the Przr is subcooled, but is being heated by the Przr heaters. The vapor space is being compressed by the excessive charging flow (& isolated letdown). This tends to maintain the Przr pressure higher than that at equilibrium conditions.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
028	Pressurizer (PZR) Level Control Malfunction	AK3.02	2.9	3.2

K/A Statement: Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions:
-Relationships between PZR pressure increase and reactor makeup/letdown imbalance

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	II.F.3	12	2
Steam Tables				
Background Information for Reactor Trip Recovery	1ES-0.1	Steps 9, 10	3	17

Facility Learning Objective:	P8197L-011 #6
Question Source:	Initial Bank question P8197L-012 036
Comments:	Question was applied to SI condition but changed to similar conditions that result from failure of level control channel. No changes to selections or correct answer

Record #	91
RO #	76
SRO #	69

Stem:

Given the following conditions on Unit 2:

- A plant calorimetric has been performed to indicate the Unit is at 15.2% power
- Power range N-41indicates 15.1%
- Power range N-42 indicates 15.4%
- Power range N-43 indicates 15.2%
- Power range N-44 indicates 15.3%
- Intermediate range N-35 indicates 7E -5 amps
- Intermediate range N-36 indicates 7E -6 amps

Which of the following conditions is indicated by these readings?

Answers/Distracters:

a. N-35 compensation voltage is too HIGH.

b. N-36 detector voltage is too LOW.

c. N-43 detector voltage is too LOW.

d. N-42 channel was incorrectly adjusted during the last calorimetric.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F	1	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- - With compensation voltage higher than normal (overcompensated) the channel reads lower than expected. If this were true then N-35 would be expected to read less than N36. It is he opposite of undercompensated so the candidate will have to consider this
b	Correct- The intermediate range to power range overlap is expected to occur at approximately 6-10E -6 amps. At 15 % power the IR indication should be approximately a decade and a half higher (~ 7E -5 amps) than at overlap. With an IR ion chamber, the higher the operating voltage on the detector, the higher the output. Since N-35 is indicating near normal, it is indicative that the voltage on N-36 is lower than expected.
c	Incorrect-- N-43 is the lowest reading PR but is acceptable both within the channel agreement and with the calorimetric.
d	Incorrect- - Adjustment of the PR is not required with the given PR output values (within limits of Calorimetric).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
033 Loss of Intermediate Range Nuclear Instrumentation	AK1.01	2.7	3.0

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation: -Effects of voltage changes on performance

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	II.A.5.d,V.A.2.f	10, 43	4

Facility Learning Objective:	P8184L-002 #2,17
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Initial Bank question P8184L-002 031. Modified to correlate to actual failure that would cause reading. All selections changed. Correct answer changed. Change some values for PR & IR output.

Record #	92
RO #	77
SRO #	70

Stem:

Given the following conditions on Unit 1:

- The Unit is at 8% power during a startup
- Intermediate Range (IR) N35 FAILS HIGH
- Immediately thereafter, Intermediate Range (IR) N36 FAILS LOW

What action is required to be taken?

Answers/Distracters:

a. BOTH IR channels Level Trip switches are **placed** in BYPASS AND power escalation can then continue above 10% power.

b. The reactor must be **placed** in HOT SHUTDOWN within one hour.

c. The reactor trips, BOTH Source Range channels immediately reenergize AND the SR High Voltage must be removed by **holding** the SR Block/Reset switches to BLOCK.

d. The reactor trips AND the Source Range channels must be manually **reenergized**.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	1-I	1	4	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is the expected action to be taken for the above conditions if power is above the P-10 setpoint (10%power) with the IR trips blocked.
b	Incorrect-- Action is to be taken to shutdown the reactor with power less than P-10 (10% power) if a single IR channel fails such that it does not result in a reactor trip. Failure of both IR channels would require entry into TS3.0.C.
c	Incorrect The reactor does trip, but the IR channels would only result in SR reenergization if both IR channels had failed low (below the P-6 setpoint).
d	Correct- At below 10% power, the IR trips are active (not blocked). Failure of either channel high will result in a IR high power trip of the reactor. Normally following a trip the IR flux will decrease and the SR instruments will be reenergized when the flux level on both IR channels falls below the P-6 setpoint (Approximately 15 minutes following a trip). With one channel failed high, the SR channels will not be energized and per direction of ES-0.1, the operator must manually energize the SR instruments.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
033 Loss of Intermediate Range Nuclear Instrumentation	AK3.02	3.6	3.9

K/A Statement:	Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: -Guidance contained in EOP for loss of intermediate-range instrumentation

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	IV.B.4, V.C	24-26, 45	4
Instrument Failure Guide	1C51.1	IR Nuclear Instrument N-35 - High	1	14
Reactor Trip Recovery	1ES-0.1	15	12	17

Facility Learning Objective:	P8184L-002 #8,10,17
Question Source:	New
Comments:	

Record #	93
RO #	
SRO #	71

Stem:

Given the following conditions on Unit 1:

- The Unit is in MODE 6.
- A single fuel assembly was DROPPED in the refueling cavity while being transported to the upender.
- The actions of D5.2 AOP1 "Damaged Fuel Assembly" have been performed.
- The Emergency Director (Shift Manager) has classified the event as an ALERT due to the radiation conditions associated with the accident.

What would be the expected containment area radiation level associated with this event?

Answers/Distracters:

a. 25 mRem/hr

b. 400 mRem/hr

c. 250 R/hr

d. 600 R/hr

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b.	1-F,P	1	3	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect– 25 Rem is not a evaluated radiation level, but is approximately ten times the highest expected background level during fuel movement (2.5 mRem/hr).
b	Correct- For an ALERT classification in a fuel handling accident, the Containment Area monitors (R-2, R-7) would read > 350 mRem/hr. At 400 mRem/hr level, the classification is correct. If the radiation levels > 200 R/hr on high range containment monitors (R-48, R-49), with ED opinion, the event should be classified as a SITE AREA EMERGENCY.
c	Incorrect– As above this is above the expected limit for declaring a SITE AREA EMERGENCY.
d	Incorrect- This level is above any evaluation criteria.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
036 Fuel Handling Incidents	AA2.03	3	3

K/A Statement:	
	Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: -Magnitude of potential radioactive release

Reference Title:	Reference #:	Section:	Page:	Rev:
Fuel Handling	P8182L-003	IV.D.1 IV.E.1	58-59, 64	4
CLASSIFICATION OF EMERGENCIES ATTACHMENT 1	F3-2	Condition 13A	44	27W

Facility Learning Objective:	P8182L-003 #12, 13
Question Source:	New
Comments:	

Record #	94
RO #	
SRO #	72

Stem:

Given the following conditions on Unit 1:

- Reactor power is 30% AND stable
- RCS Tavg is 551°F
- Pressurizer pressure is 2230 psig
- Pressurizer level is 25%
- Pressurizer Level Control Transfer Switch is in the NORMAL 2-3 position
- The speed sensor for the running Charging Pump FAILS HIGH

What automatic action(s) will occur over time as a result of this failure?
(assume NO operator action taken)

Answers/Distracters:

a. Pressurizer level will rise to 50% AND stabilize.

b. Pressurizer level will drop to 21% AND stabilize.

c. Letdown will isolate AND the reactor will trip on high Pressurizer level.

d. Backup heaters will energize AND the reactor will trip on high Pressurizer pressure.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-DR,RI	1	3	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- This is the maximum Pressurizer control level corresponding to failed temperature input (Tref) of 580°F.
b	Incorrect- This is the normal zero-power Pressurizer level setpoint.
c	Correct- When the speed sensor fails high it provided input that the pump is running faster than the demanded speed. It would then attempt to continue to slow the pump acting in the same manner as if the controlling level channel high fails high, in that charging flow falls to minimum. At 14.8% level, letdown isolates charging continues at minimum (26 gpm for 2 pumps running) and Przr level rises to high level trip setpoint.
d	Incorrect- This may occur if the output fails high. The Pressurizer level would rise and at 10% above program level, the backup heaters (Group D and Group E) would be energized. However the pressure control system would be able to control the pressure by opening spray valves and/or PORVs as necessary. Also the pressure rise would be somewhat limited by the insurge of the cooler RCS water as level rises.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
028 Pressurizer (PZR) Level Control Malfunction	AK2.02	2.6	2.7

K/A Statement:	Emergency Procedures / Plan -Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.

Reference Title:	Reference #:	Section:	Page:	Rev:
Pressurizer Level Control System	P8170L-006	IV.A.3.	13	3
REACTOR CONTROL SYSTEM	B7	3.4.2	22-24	4
Interlock Logic Diagram Chemical Volume and Control System	NF-40784-4			J
Interlock Logic Diagram Pressurizer System Unit 1	NF-40780-2			H

Facility Learning Objective:	P8170L-006 #6,7
Question Source:	New
Comments:	Replaces 036 2.4.8 (SRO only). No 036 APE KA selected on replacement outline. KA selected from previously suppressed KA that was on replacement outline.

Record #	95
RO #	
SRO #	73

Stem:

Given the following conditions on Unit 1:

- The Unit is in MODE 1 at normal operating pressure and temperature
- The operator has reported INCREASED charging pump speed AND charging flow
- Pressurizer level is 20% AND slowly trending down
- VCT level is 24% AND trending down
- R-15, Condenser Air Ejector Gas monitor AND R-19, SG Blowdown Liquid monitor show an increasing trend

What is the first action that requires the NRC Resident Inspector to be notified?

Answers/Distracters:

a. Following entry into 1C4 AOP2 "Steam Generator Tube Leak".

b. Following determination of the leak rate AND Action Level 2 steps for SG tube leakage are initiated.

c. When the leak rate is determined to have EXCEEDED 10 gpm.

d. When the leak rate is determined to have EXCEEDED the charging pump capacity AND the event has been classified.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a.	1-P	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct- Notification of GSPO and NRC Resident Inspector is required for entry into 1C4 AOP2 per SWI O-28 appendix A
b	Incorrect- The notification steps in 1C4 AOP2 are prior to the determination of actual leakrate
c	Incorrect- The notification steps in 1C4 AOP2 are prior to the determination of actual leakrate
d	Incorrect-The conditions given indicate a SG tube leak that is less than the capacity of the charging pumps, therefore entry into 1C4 AOP2 would occur prior to the leakage exceeding charging pump capacity. Which would require a manual reactor trip.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
037 Steam Generator (S/G) Tube Leak	2.4.30	2.2	3.6

K/A Statement:	Emergency Procedures / Plan: -Knowledge of which events related to system operations/status should be reported to outside agencies.

Reference Title:	Reference #:	Section:	Page:	Rev:
SG Tube Leak and AFW AOPs	P8140L-135	III.A.2.b	8	0
Steam Generator Tube Leak	1C4 AOP2	2.4.6	5	10

Facility Learning Objective:	P8140L-135 #1 P9150L-011 #8
Question Source:	New
Comments:	

Record #	96
RO #	78
SRO #	74

Stem:

Given the following conditions on Unit 1:

- A reactor trip and Safety Injection has occurred
- The operators have completed the actions of 1E-0 "Reactor Trip Or Safety Injection" and transitioned to 1E-3 "Steam Generator Tube Rupture" based on radiation indications for 11 SG
- SI has been reset
- RCS pressure is 1500 psig
- RCS core exit temperature is 535°F
- 11 SG pressure is 1050 psig
- 12 SG pressure is 875 psig
- RCS cooldown has been initiated to 505°F as directed by 1E-3 using12 SG and condenser steam dump

If a loss of offsite power occurs to Bus 15 at this time, what would the RHR Pump control switch indications be following sequencing of loads?

Answers/Distracters:

	<u>11 RHR Pump</u>	<u>12 RHR Pump</u>
a.	RED light lit	RED light lit
b.	RED light lit	GREEN light lit
c.	GREEN light lit	RED light lit
d.	GREEN light lit	GREEN light lit

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	2-RI	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This would only be expected if SI had not been reset and the 11 RHR pump restarted.
b	Incorrect- This would be the case if RHR Pump restart on the loss of power sequencing, after being stopped. The RHR Pump start on SI only, if previously stopped
c	Incorrect- This would be expected if the RHR pumps had been running and the loss of load sequencer (as it normally does) does not start the RHR Pump. This is incorrect if the RHR Pump is running when the power loss occurs.
d	Correct- Prior to initiating the RCS cooldown SI is reset and the RHR pumps are stopped if RCS pressure is > 250 psig. SO the RHR pumps are stopped. A loss of power to Bus 15 starts loss of load sequencing which does not include the RHR pumps if they are not running. Therefore no change is expected to exist for RHR Pump indication.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
038	Steam Generator Tube Rupture (SGTR)	EA1.22	2.7	2.6

K/A Statement:	Ability to operate and/or monitor the following as they apply to a SGTR: -RHR pump operation and indications.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-3 Series Review	P8197L-13A	X.D.1.d.1.g	68	2
Residual Heat Removal	P8180L-003	IV.A.8.b	19	3
Steam Generator Tube Rupture	1E-3	Step 13.C	8	15

Facility Learning Objective:	P8197L-13A #18, P8180L-003 #6.a, 7
Question Source:	New
Comments:	

Record #	**97**
RO #	
SRO #	**75**

Stem:

(see reference)

Given the following conditions on Unit 1:

- The reactor tripped from 99.5% power due to a generator trip
- Buses 11 and 12 have FAILED TO TRANSFER to 1R transformer following the trip AND power cannot be restored to either bus for 4 days
- All CST levels are at 100,000 gallons AND makeup water is available
- All plant parameters stabilize at no-load conditions without SI actuation
- Following completion of the IMMEDIATE ACTION steps of 1E-0 "Reactor Trip Or Safety Injection" entry is made into 1ES-0.1 "Reactor Trip Recovery"

Which of the following describes the direction provided by the Unit 1 Shift Supervisor for controlling the plant over the period busses 11 and 12 remain out of service?

Answers/Distracters:

a. **Transition** as directed to 1C1.2 "Unit 1 Startup Procedure" maintaining stable no-load conditions.

b. **Remain** in 1ES-0.1 "Reactor Trip Recovery" maintaining stable no-load conditions until at least one RCP is available.

c. **Transition** as directed to 1C1.3 "Unit 1 Shutdown" AND cooldown the RCS to 340°F.

d. **Transition** as directed to 1ES-0.3A "Natural Circulation Cooldown With CRDM Fans" AND cooldown the RCS below 350°F.

Answer:	***LOK***	***Tier:***	***LOD***	***RO Group:***	***SRO Group:***	***Facility:***	***Exam Date:***
d	3-SPR, PEO	1	4	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This option is available if RCS cooldown is not required and stable no-load conditions are to be maintained.
b	Incorrect- This is the desired condition. However the given conditions ensure that RCPs are not available prior to the expected completion of ES-0.1. ES-0.1 does provide for transition depending on given conditions. Since cooldown is required by Tech Specs and NO RCPs are available, then transitions and cooldown per 1ES-0.3A is the expected action.
c	Incorrect- This option would be proper if at lest one RCP was available and a cooldown is required (Tech Spec 3.1.A.1.b.(2) for one RCP operable).
d	Correct- The directions to be taken for operating without forced RCS flow is ultimately determined by Technical Specifications. ES-0.1 has each of the three options available for procedural guidance (transition to 1C1.2, transition to 1C1.3, transition to 1ES-0.3A) depending on the given condition. Tech Spec 3.1.A.1.b requires both RCS loops being operable. With both loops inoperable, immediate action is directed to establish plant conditions to minimize the effects and if one RCP is not restored within 72 hours a cooldown to less than 350°F within the following 12 hours. With the given bus availability, the RCPs will be unavailable for at least the allowed time and the proper action is initiate cooldown. If an RCP were available, the transition would be to 1C1.3 if cooldown is required and to 1C1.2 if cooldown is not required.

K/A System/Evolution:		***K/A #:***	***KAVRO***	***KAVSRO***
E09	Natural Circulation Operations	EA2.1	3.1	3.8

K/A Statement:	Ability to determine and interpret the following as they apply to the Natural Circulation Operations: -Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Reference Title:	***Reference #:***	***Section:***	***Page:***	***Rev:***
E-O Review	P8197L-011	VI.C	18-19	2
Background Information for Reactor Trip Recovery	1ES-0.1	Step 23	5	17
Prairie Island Technical Specifications		3.1.A.b.(3)	TS.3.1-1	157

Facility Learning Objective:	P8197L-011 #11,15
Question Source:	Significantly modified from NRC Exam Bank
Comments:	Material Required for Examination-TS 3.1-1 (page TS.3.1-1 only) Vogtle 12/1999 SRO NRC exam. Question 19. Conditions changed to reflect specific plant conditions. Tech Spec evaluation added to determine required cooldown actions,; therefore correct answer changed.

Record #	98
RO #	79
SRO #	76

Stem:

Given the following conditions on Unit 1:

- A loss of all feedwater has occurred
- RCS temperature is 570°F and stable
- RCS pressure is 2290 psig
- 11 SG wide range level is 6%
- Condensate flow has just been established

Which of the following statements describes the possible result of initiating full feed flow to the 11 SG?

Answers/Distracters:

a. Steam Generator tube failure could occur due to caustic stress corrosion.

b. An overcooling event could occur due to excessive steaming.

c. The RCS could be subjected to a Pressurized Thermal Shock (PTS) event.

d. Steam Generator component degradation could occur due to significant thermal stresses.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d.	1-F,P	1	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Caustic stress corrosion is a concern for the RCS during SI but not for feeding a dry SG.
b	Incorrect- Given the conditions, overcooling and PTS are not a concern for controlling the feedwater flow. However these may be considered since cooling will occur when the flow is initiated.
c	Incorrect- Given the conditions, overcooling and PTS are not a concern for controlling the feedwater flow. However these may be considered since cooling will occur when the flow is initiated.
d	Correct- A SG level of < 9% may indicate the SG has dried out. For this condition, establishing feed flow at a rate in excess of 100 gpm may cause significant thermal stress to SG components.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
054 Loss of Main Feedwater (MFW)	AK1.02	3.6	4.2

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):
-Effects of feedwater introduction on dry S/G

Reference Title:	Reference #:	Section:	Page:	Rev:
F/FR Review	P8197L-014	IV.D.2.c	29	2
Background Information for 1FR-H.1 Response to Loss of Secondary Heat Sink	1FR-H.1	NOTE for Step 21	9	12

Facility Learning Objective:	P8197L-014 #19
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8197L-014 053. Modified premise layout to give condition that represents a hot, dry SG with conditions requiring establishing feed. Question did change and selections remain the same.

Record #	**99**
RO #	**80**
SRO #	

Stem:

Given the following conditions on Unit 1 AND Unit 2

- A Total Loss of All AC Power Event has occurred

How is operation of the valves associated with the RCS affected when the Instrument Air header depressurizes?

Answers/Distracters:

a. Reactor Head Vent valves CANNOT be OPENED.

b. Pressurizer PORV's will NOT OPEN on demand.

c. RCP No. 1 Seal Leakoff valves CANNOT be RE-OPENED.

d. Normal Pressurizer Spray valves will NOT OPEN on demand.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
d	1-F,I	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- The Reactor Head Vent valves , SV-37037 and SV-37038 are solenoid valves powered from Safeguards DC power and are not immediately affected by the Loss of AC power or Instr Air.
b	Incorrect- PORVs require air to open; however, the valves are supplied with a backup air accumulator. This allows operation for up to approximately 15 times
c	Incorrect- RCP No.1 Seal leakoff valves fail open on loss of air. The valves cannot be closed.
d	Correct- The Przr spray valves fail close on a loss of air and have no backup supply.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
055 Loss of Offsite and Onsite Power (Station Blackout)	EA2.01	3.4	3.7

K/A Statement:	Ability to determine and interpret the following as they apply to a Station Blackout: -Existing valve positioning on a loss of instrument air system

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
E-O Review	P8197L-011	V.E.5	26	2
Instrument and Station Air	P8178L-005	V.C.1	21	2
LOSS OF INSTRUMENT AIR	C34 AOP1	Attachment A, 2.A	9	10

Facility Learning Objective:	P8197L-011 # F.19 P8178L-005 #8
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Initial Bank question P8178L-005 012. Conditions changed to fit Loss of AC power, resulting in loss of pressure in IA header, and two selections changed to valves related to RCS isolation/control,

Record #	100
RO #	
SRO #	77

Stem:

Given the following conditions on Unit 1 and Unit 2:

- A Total Loss of All AC Power Event has occurred
- The actions of ECA-0.0 "Loss of All Safeguards AC Power" are being performed
- RCS Core Exit Thermocouple temperature is 500°F
- RCS pressure is 1950 psig
- Attempts to restore bus power are underway but not immediately expected
- RCS cooldown and depressurization has been initiated by depressurizing intact SGs.

What is the cooldown rate restriction AND its basis?

Answers/Distracters:

a. The RCS cooldown rate SHALL NOT exceed 10°F in any one-hour period to minimize the formation of voids in the reactor vessel head.

b. The RCS cooldown rate SHALL NOT exceed 25°F in any one-hour period to prevent the formation of voids in the reactor vessel head.

c. The RCS cooldown rate SHALL NOT exceed 100°F in any one-hour period to minimize the thermal stresses across the reactor vessel wall and prevent crack propagation.

d. The RCS cooldown rate MAY exceed 100°F in any one-hour period to allow depressurization of the SGs to designated pressure as quickly as possible to minimize RCS inventory loss.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-B,F	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This would be the limit for natural circulation cooldown if the CRDM fans are NOT available (as might occur on loss of power to non-safeguards buses).
b	Incorrect- This is the limit for cooldown based on natural circulation with CRDM fans available to provide additional head cooling.
c	Incorrect- This is the basic RCS cooldown limit as related to Technical Specifications (PTLR, Rev. 1) based on the ASME limits associated with limiting crack propagation in the vessel wall.
d	Correct-] With a loss of all AC power, the RCS is in a natural circulation condition. In most instances when natural circulation is underway, cooldown restrictions exist based on available equipment. In most other instances the Technical Specification cooldown limit is applicable. However, in the event of a loss of AC power, the cooldown rate is less important than the requirement to depressurize the RCS in order to preserve RCS inventory. In this special case there are no limits on the cooldown rate (other than the equipment operational limitations).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
055	Loss of Offsite and Onsite Power (Station Blackout)	EA2.02	4.4	4.6

K/A Statement:	Ability to determine and interpret the following as they apply to a Station Blackout: -RCS core cooling through natural circulation cooling to S/G cooling

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	V.C.3, V.E.3.b.2	23-24,25	2
Background Information for 1ECA-0.0 Loss of All Safeguards AC Power	1ECA-0.0	Bases 1st NOTE, Step 28	8	14

Facility Learning Objective:	P8197L-011 # F.20
Question Source:	Significantly modified from Facility Exam Bank
Comments:	Reqaul Bank Part B question P8197L-011 004. Original question was written to address natural circulation cooldown conditions with AC power available. Changed to Loss of all AC and changed 2 selections, correct answer to conditions provided by event.

Record #	101
RO #	81
SRO #	78

Stem:

When performing the actions of ECA-0.0 "Loss of All Safeguards AC Power" during a Total Loss of All AC Power Event, why does the procedure direct the operator to reset the SI signal?

Answers/Distracters:

a. To ALLOW for another start attempt of the Emergency Diesel Generators.

b. To ALLOW for realigning of equipment needed to respond to the accident.

c. To PREVENT automatic loading of the power supply once power restoration has been completed.

d. To ALLOW the operator to transition to ECA-0.1 "Loss of All AC Power Recovery Without SI Required" REGARDLESS of RCS conditions.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-B,P	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This action does not affect the operational status of the DGs. These may be taken to LOCAL control for starting, with the SI signal still present
b	Incorrect-- While resetting the SI signal allows the OPERATOR to manually load SI equipment as directed in the recovery procedure(s), it does not allow at this time realigning equipment, as AC power is still not available for the affected valves, pumps and other equipment. The operator cannot realign equipment.
c	Correct- This action aids in preventing the automatic loading of equipment on the Safeguards Buses once power is restored. Since the source of power may be a DG loading is a major concern.
d	Incorrect- This is determined by plant conditions (RCS subcooling and Przr level) at a time following power restoration to at least one Safeguards Bus and is evaluated at that time

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
055 Loss of Offsite and Onsite Power (Station Blackout)	EK3.02	4.3	4.6

K/A Statement:	Knowledge of the reasons for the following responses as they apply to the Station Blackout: -Actions contained in EOP for loss of offsite and onsite power

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	V.E.3	24-25	2
Background Information for 1ECA-0.0 Loss of All Safeguards AC Power	1ECA-0.0	Step 30	11	14

Facility Learning Objective:	P8197L-011 # F.20
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8197L-011 010.

Record #	102
RO #	82
SRO #	

Stem:

Given the following conditions on Unit 2:

- A loss of offsite power had occurred
- Bus 26 is powered from Diesel Generator D6
- The normal power source to Bus 26 is NOT available
- The alternate power source to Bus 26 has been restored
- The Shift Supervisor has directed that Bus 26 be restored to its offsite power source

What position would the operator need to place Bus 26 Synch Selector switch in to allow this transfer?

Answers/Distracters:

a. **CT11**

b. **CT12**

c. **D6**

d. **2RY**

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-F	1	2	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is not a potential source of power to a Unit 2 bus, unless the bus is cross-tied to Bus 16 which may be powered from CT11 as its normal source.
b	Incorrect- CT12 is the normal power source to Bus 26 but is listed as unavailable.
c	Incorrect- This is the position normally used by operators which allows D6 to be paralleled to Bus 26 for testing. It is not used for DG shutdown.
d	Correct- The alternate source of power to Bus 26 is 2RY. In order to align the bus to this power source the synchronization switch must be aligned to that source, 2RY.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
056 Loss of Offsite Power	AA1.02	4.0	3.9

K/A Statement:	Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: -ESF bus synchronization select switch to close bus tie breakers

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguards 4160V & 480V Electrical Dist	P8186L-008	IV.D.1	14-15	4
D5/D6 DIESEL GENERATORS	2C20.7	5.7.4.D	95	16

Facility Learning Objective:	P8186L-008 #4.a, 5.b
Question Source:	New
Comments:	

Record #	103
RO #	
SRO #	79

Stem:

Given the following conditions on Unit 1:

- The Unit is in MODE 4 at 250°F
- 11 Inverter has FAILED
- Instrument Panel 111 is powered from Interruptible Panel 117
- 14 Inverter experiences an internal POWER FAILURE

In performing the actions of 1C20.8 AOP1 "Abnormal Operation, Instrument AC Inverters", why is it PROHIBITED to also power Instrument Panel 114 from Panel 117 simultaneously?

Answers/Distracters:

a. To assure Panel 117 is not overloaded.

b. To prevent paralleling BOTH trains of battery chargers with each other.

c. To assure adequate power would be available to start both Diesel Generators.

d. To prevent a single fault condition from defeating MORE THAN ONE set of a plant protection system features.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-B	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-Panel 117 is supplied from MCC 1AC1 via 11 aux transfer or may be supplied from panel 217. The panel is capable of supplying normal loads for two instrument buses without affecting load carrying capacity. This can be considered since Proc. 1C20.8 contains information concerning evaluation of loads if panel 117 is powered from panel 217, and panel 217 is supplying alternate power to an instrument bus.
b	Incorrect-The battery chargers are not crosstied since the DC input to the inverter which is supplied by the battery charger is unaffected by the transfer of power source to panel 117. The crosstie to inverter output is prevented by the mechanical interlock at the instrument bus which prevents crosstieing inverter supply and panel 117 supply to the instrument bus.
c	Incorrect-DG starting is controlled by DC power (starting solenoids) which is unaffected by the transfer to panel 117, as detailed above.
d	Correct- The arrangement of the auxiliary power sources and equipment, including the Technical Specification limits on power source for the Instrument Buses (Inverter Operability), assures that no single fault condition will deactivate more than one redundant set of safeguard equipment items in one reactor and will therefore not result in failure of the plant protection system to respond adequately to a LOCA. In this case both Channel I and Channel IV would be affected and could be deactivated by a loss of power to Panel 117.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
057	Loss of Vital AC Electrical Instrument Bus	2.4.11	3.4	3.6

K/A Statement:	Emergency Procedures / Plan -Knowledge of abnormal condition procedures.

Reference Title:	Reference #:	Section:	Page:	Rev:
Safeguard Dist. 120 V AC Instrumentation	P8186L-015	IV.C.3.f, VI.A	16-17	1
ABNORMAL OPERATION, INSTRUMENT AC INVERTERS	1C20.8 AOP1	2.4.5.C	6	6
INSTRUMENT AC DISTRIBUTION SYSTEM	1C20.8	4.2	3	14

Facility Learning Objective:	P8186L-015 #7,9
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8186L-015 005. Premise changed to give conditions (as compared to above COLD SD) and question changed to direct specific answer. No changes to selections.

Record #	104
RO #	
SRO #	80

Stem:

Given the following conditions on Unit 1:

- The Unit is at 100% power
- The DC electrical system is aligned for normal at-power operations.
- A breaker FAULT on the MCC breaker supplying 11 Battery Charger has caused it to TRIP open.

Which of the following states the LCO action required for this condition?

Answers/Distracters:

a. Power must be RESTORED to 11 Battery Charger within 8 hours.

b. BOTH Instrument Panels 111 AND 113 must be ALIGNED to the Inverter bypass source within 8 hours

c. Within one hour power reduction must be INITIATED AND the Unit placed in at least HOT SHUTDOWN within the next 6 hours.

d. Diesel Generator D-1 is inoperable, D-2 must be RUN within the next 24 hours AND power RESTORED to 11 Battery Charger within 7 days.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-F,P	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct- The LCO for DC power states "Both batteries with their associated chargers and both d-c safeguards system shall be operable." (with the reactor critical). The associated ACTION states, "One battery charger may be inoperable for 8 hours provided: (a) its associated battery is operable, (b) its redundant counterpart [12 Charger] is verified operable, and (c) the diesel generator and equipment associated with its counterpart are operable."
b	Incorrect- The instrument buses are affected; however, the are still operable and by TS may be aligned to bypass source for up to 8 hours. The buses must be transferred before 8 hours or the 11 Battery will not last for 8 hours (as directed in AOP).
c	Incorrect- This may be considered as this is TS 3.0.C action for a condition not allowed by TS or covered in individual TS actions.
d	Incorrect- The DG operability is addressed in the action statement for the battery charger inoperability. The associated action statement is applicable to inoperable DG.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
058	Loss of DC Power	2.2.22	3.4	4.1

K/A Statement:	Equipment Control -Knowledge of limiting conditions for operations and safety limits.

Reference Title:	Reference #:	Section:	Page:	Rev:
DC Distribution	P8186L-005	IV.A.4.a.7	10-11	2
FAILURE OF 11 BATTERY CHARGER	1C20.9 AOP3	2.4.1	3	4
Prairie Island Technical Specifications		3.7.B.7	TS.3.7 -3	160

Facility Learning Objective:	P8186L-005 #12
Question Source:	
Comments:	PI SRO NRC Exam 5/15/2000, question 56. Modified to convert from source of DC power to TS action associated with loss of Battery Charger. All selections changed.

Record #	105
RO #	83
SRO #	81

Stem:

Which of the following radiation monitors could STOP a waste discharge if a HIGH alarm condition exists?

Answers/Distracters:

a. **R-21**, Circulating Water Discharge Monitor

b. **1R-15**, Unit Condenser Air Ejector Gas Monitor

c. **2R-19**, Unit 2 Steam Generator Blowdown Monitor

d. **R-16** Containment Fan Coils Cooling Water Discharge Monitor

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-I	1	2	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The Circ Water Discharge monitor provides alarm function only. Operator action is required to determine the source of release and to terminate.
b	Incorrect- This is indicative of a potential primary to secondary release and possible environmental release (gas) vial the condenser off gas system. However, no automatic action is associated with the alarm. Operator action is required to stop release, including alignment of the off-gas system & turbine ventilation, and possible alignment of SG blowdown.
c	Correct- The SG blowdown monitor closes the SGB to 21 SGB Flash Tank valves CV-31610, CV-31611 and 21 Flash Tank Discharge to River CV-31607, isolating any potential release to the environment.
d	Incorrect- Monitors CTMT fan coils #12, 14, 22, and 24 cooling water for inleakage during a LOCA when CTMT pressure exceeds cooling water pressure with possible indication of unmonitored release. However, as this is associated with an accident condition, no automatic actions exist. The operator is directed to C35 AOP4 if accident condition does not exist, and is directed to isolate individual CFCUs cooling water discharge valves sequentially to determine source and terminate leak.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
059 Accidental Liquid Radwaste Release	AA1.01	3.5	3.5

K/A Statement:	Ability to operate and/or monitor the following as they apply to the Accidental Liquid Radwaste Release: -Radioactive-liquid monitor

Reference Title:	Reference #:	Section:	Page:	Rev:
Radiation Monitoring System	P8182L-002	V.D.4.d	28	4
Alarm Response Procedures	C47047	47007 2R-19	1	23

Facility Learning Objective:	P8182L-002 #5
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8182L-002 005. Changed two selections to divide choices between Units and Common detectors. Also selections are related to liquid systems (indirectly for 1R-15).

Record #	106
RO #	84
SRO #	82

Stem:

(see reference)

Using the reference provided, identify the radiation monitor first affected if the inservice Gas Decay Tank (127) relief valve lifts?

Answers/Distracters:

a. **R-35**, Rad Waste Building monitor

b. **R-41**, Waste Gas High Level monitor

c. **2R-22**, Shield Building vent stack monitor

d. **1R37**, Aux Bldg vent stack train A monitor

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-RI	1	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The RW Building monitor provides continuous monitoring of gas activity in the RW (rad waste) Bldg effluent but is not connected to the Waste Gas system, but may be considered since the high level loop gas decay tanks relieve to the HUTs header.
b	Incorrect- This monitors suction header to gas compressors. This may be considered.
c	Correct- The low-level loop gas decay tanks relieve to a common header that is directed to the Unit 2 Shield Bldg Vent System. In this case the first monitor that samples from this line is 2R-22.
d	Incorrect- This monitor samples Unit 1 Auxiliary Building exhaust and provides monitoring and auto actions to limit offsite releases within 10CFR 20. This monitor could be affected if the normal release path is operated.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
060 Accidental Gaseous Radwaste Release	AK2.02	2.7	3.1

K/A Statement:	Knowledge of the interrelations between the Accidental Gaseous Radwaste Release and the following: -Auxiliary building ventilation system

Reference Title:	Reference #:	Section:	Page:	Rev:
Rad Waste - Waste Gas	P8182L-001C	V.B.1	13	2
Radiation Monitoring System	P8182L-002	III.D.2	13	4
Flow Diagram Waste Disposal System Unit 1& 2	X-HIAW-1-124			S

Facility Learning Objective:	P8182L-001C #8
Question Source:	New
Comments:	Material Required for Examination Flow Diagrams X-Hiaw 1-124

Record #	107
RO #	85
SRO #	83

Stem:

Given the following conditions on Unit 1:

- The Unit is in MODE 2 with reactor startup in progress
- Reactor power has been stabilized at 1 x 10-8 amps and critical data has been recorded
- I & C is troubleshooting the erratic indication that has developed on Intermediate Range channel N-35
- At the NIS rack, the Level Trip switch for N-35 has been placed in BYPASS

What occurs if the technician then pulls one of the Control Power fuses for N-35?

Answers/Distracters:

a. The reactor trips on Intermediate Range High Flux Level.

b. The power increase is limited to 2% power until N-35 is restored.

c. BOTH Source Range NIS channels unblock resulting in a reactor trip.

d. Control Rod withdrawal is blocked but the rods may be inserted in manual.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-DR,RI	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct- The removal of control power the input to the reactor trip actuates. The only time this input may be blocked is if reactor power is above P-10 setpoint AND the IR TRIP BLOCK control switches on the control board have been taken to BLOCK. This blocks the input to RPS from the IR NIS preventing a loss of control power from initiating the trip.
b	Incorrect-- By Technical Specification if one channel of IR NIS is inoperable (as it is in this case) and the reactor does not trip, power is limited to remaining below the P-10 setpoint (10% power). This may be considered since 2% power is the MODE 1 entry level.
c	Incorrect- This may be considered since the input from IR NIS affects the operation of the SR NIS. The SR trip and HV power was blocked when the IR NIS exceeded the P-6 setpoint (1x10(-10) amps). The loss of control power to one channel does not affect the SR since both IR channels must be below the P-6 setpoint to automatically reinstate the SR NIS. The SR NIS is expected to unblock following the decrease in reactor power FOLLOWING the reactor trip, but not causing the reactor trip.
d	Incorrect- This may be considered since the Level Trip also affects the IR Rod Stop. However since the reactor trips, this result is not possible (Rod motion not allowed with trip breakers open).

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
033 Loss of Intermediate Range Nuclear Instrumentation	AA1.02	3.0	3.1

K/A Statement: Ability to operate and/or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation:
-Level trip bypass

Reference Title:	Reference #:	Section:	Page:	Rev:
Nuclear Instrumentation System	P8184L-002	IV.B.2 IV.B.3.b).(11)	20-21 23	4
Nuclear Instr & Manual Trip Signals	XHIAW-1-237			A

Facility Learning Objective:	P8184L-002 #10,11
Question Source:	New
Comments:	

Record #	108
RO #	
SRO #	84

Stem:

Given the following conditions on Unit 1:

- The Unit is in refueling operations
- Train B RHR is operating
- Refueling Cavity level is at 750' 2"

What is the basis that allows for inoperability of the 11 RHR Pump without compensatory action in this situation?

Answers/Distracters:

a. The effect of an inadvertent dilution is minimal.

b. Operation of two pumps increases the likelihood of vortexing at the RHR pumps suction.

c. The available heat sink allows sufficient time to initiate alternate core cooling if RHR is lost.

d. It allows for ease in handling fuel assembly removal and placement in certain core locations.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-B	1	3	3	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- Dilution is a concern and the basis states that the RHR pump is used to maintain a uniform boron concentration. This is not the reason for allowing one pump out of service, but a reason for having at least one operable when boron changes are in progress.
b	Incorrect- This is a consideration only for operation in reduced RCS inventory condition (Refueling level > 3' below reactor vessel flange).
c	Correct- The basis for allowing water level to be lowered below 20 feet (top of RCCA drive shafts) with only one train of RHR operating is "the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core...".
d	Incorrect- Both RHR Pumps may be stopped during core loading for up to one hour to facilitate movement of fuel or core components. However the required loop(s) must remain operable.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
025 Loss of Residual Heat Removal System (RHRS)	2.2.25	2.5	3.7

K/A Statement:	Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Reference Title:	Reference #:	Section:	Page:	Rev:
Residual Heat Removal	P8180L-003	VII.	47	3
Fuel Handling	P8182L-003	V.B.2.a).(6) V.C.2.b).(7)	68 73	4
Prairie Island Technical Specifications Bases		3.8 Bases	B.3.8-2	130

Facility Learning Objective:	P8180L-003 #11 P8182L-003 #18
Question Source:	New
Comments:	Replaces 065 AA2.04 (SRO only). No acceptable KA was selected on initial replacement outline(SRO-Only). KA selected from an additional SRO-level generated random outline and selected from a previously unselected APE.

Record #	109
RO #	
SRO #	85

Stem:

Given the following conditions:

- A fire in the Relay Room causes a reactor trip
- The crew enters 1E-0, "Reactor Trip or Safety Injection"
- The fire then makes the Control Room uninhabitable and the decision is made to evacuate the Control Room.

What should be done regarding the performance of actions as directed by 1E-0?

Answers/Distracters:

a. **Complete** immediate actions of 1E-0.

b. **Exit** 1E-0 AND implement F5 App B, "Control Room Evacuation (Fire)".

c. **Enter** F5 App B AND perform 1E-0 actions in parallel.

d. **Continue** with 1E-0, substituting LOCAL actions for Control Room actions.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-FP	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:	
a	Incorrect-- Although reactor trip and turbine trip is required prior to control room evacuation, there is no need to check for Safeguards Bus power/enter ECA-0.0 (if not powered); Nor is there any need to check for SI actuation (non-accident situation).
b	Correct- The only applicable task is to perform the actions of F5 Appendix B. This procedure is modeled to take those actions required to ensure safe plant shutdown and stabilize the plant conditions as required from local locations. The following are the assumed conditions prior to evacuation: 1) From the Control Room the reactors and turbines are tripped, and the MSIVs and PORV block valves are CLOSED. 2) Normal letdown and charging were in service prior to the fire.
c & d	Incorrect-- As stated, F5 App B contains all the necessary actions necessary to achieve and maintain hot shutdown and to cooldown to cold shutdown. E-0 should not be performed. Each of these may be considered since in most other circumstances E-0 would be performed and any AOPs performed in conjunction with EOP performance.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
067	Plant Fire on Site	2.4.6	3.1	4.0

K/A Statement:	Emergency Procedures / Plan -Knowledge symptom based EOP mitigation strategies.

Reference Title:	Reference #:	Section:	Page:	Rev:
F5 Appendix B/C Review	P8197L-009	III.B	10	3
CONTROL ROOM EVACUATION (FIRE)	F5 APPENDIX B	3.0,ATT. C	5,20	22

Facility Learning Objective:	P8197L-009 #1
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8197L-009 028.

Record #	110
RO #	86
SRO #	

Stem:

Given the following conditions for both Units:

- Work is being performed on the Fire Protection Main 10" loop
- The control room operator has placed 121 Fire Pump control switch in PULLOUT
- The APEO has placed the local control switch for 122 Fire Pump in OFF
- Miss-operation of a valve causes the fire protection system header pressure to decrease to 70 psig momentarily

What actions would be required to restore the Fire Protection System to its previous status? **(Assume the miss-operated valve was immediately restored to its normal position)**

Answers/Distracters:

a. 121 Fire Pump must be **stopped** from the control room by placing its control switch in STOP, AND 122 Fire Pump must be **stopped** by taking its local control switch to AUTO AND THEN to OFF.

b. BOTH 121 AND 122 Fire Pumps must be **stopped** by depressing their respective local STOP pushbuttons.

c. ONLY 122 Fire Pump must be **stopped** by taking its local control switch to AUTO AND THEN to OFF.

d. ONLY 121 Fire Pump must be **stopped** by depressing its local STOP pushbutton.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	1-I,P	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This is the procedure directed action for stopping the pumps if they had been manually started. Both pumps do not start from the auto signal.
b	Incorrect-- This is only true for the 121 Pump. The 122 pump does not start and its control does not use pushbuttons.
c	Incorrect-- This may be considered the reverse of actual conditions. If the 122 Pump response is confused with the 121 pump response then this would be the expected response.
d	Correct- With the control room control for the 121 Fire Pump in PULLOUT, the pump will still start if an automatic start signal occurs (low header pressure of 100 psig). To prevent the auto start, procedure directs that power supply breaker must be tripped. The 121 Pump, if auto started, can only be stopped by depressing the local STOP pushbutton for the pump. With the local control for the 122 Pump in OFF, the pump will not respond to any start signal.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
067	Plant Fire on Site	2.1.23	3.9	4.0

K/A Statement:	Conduct Of Operations -Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Reference Title:	Reference #:	Section:	Page:	Rev:
Fire Detection and Protection System	P8178L-002	III.B.1	11	2
FIRE PROTECTION & DETECTION SYSTEMS	C31	3.1	4	29

Facility Learning Objective:	P8178L-002 #2,6
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8178L-002 003. Change question from direct auto start and action to stop to premise that gives condition that gives auto start for both pumps and conditions that limit starts. PI has feature that allows 121 mdfp to auto start even if its control switch in the control room is in pull out.

Record #	111
RO #	87
SRO #	86

Stem:

During a control room evacuation, why are BOTH Feedwater Pumps AND all but ONE Condensate Pump tripped as directed by 1C1.3 AOP1, "Shutdown From Outside The Control Room - Unit 1"?

Answers/Distracters:

a. To MAINTAIN minimum flow in the feedwater header AND allow cooldown of the main condenser.

b. To MAINTAIN the main condenser vacuum allowing cooldown using the Condenser Steam Dumps.

c. To PREVENT overfeeding of SGs as flow is established using the Bypass Feedwater Control valves.

d. To PROVIDE adequate feedflow to maintain core cooling for the first 30 minutes following the reactor trip.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-B	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- The only flowpath for feedwater is via Recirc Valves (Feed Reg Valves & Bypasses closed), and no controls are available for the feedwater system from HSD panel. So all Pumps except one Condensate Pump are tripped to maintain Cond Hdr Pressurized, CD pump Seal Water, minimize Recirc flow, and the Cond Pump Recirc CV is verified open to maintain the flow path to allow cooling of the pump and main condenser.
b	Incorrect- the Circ Water system and Air Removal system are required to maintain condenser vacuum. The SG PORVS have controls at the HSP and are used to provide SG cooling. Steam dumps have no controls at the hot shutdown panel.
c	Incorrect-- Bypass FW CVs are closed, not used for flow control during shutdown outside the control room. AFW system is used to provide makeup flow to SGs. With Condensate Pump running the SG pressure would have to be reduced to approx. 400 psig to establish feed flow.
d	Incorrect- The AFW system is designed such that the 200 gpm output of each pump is capable of removing equivalent to a decay heat rate of 1.69% full power. This is rapidly achieved within a few minutes following a reactor trip. But since the Pumps are not rated for several % power, the operator may confuse this bases for AFW Pump capability with need to maintain FW flow.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
068 Control Room Evacuation	AK3.08	3.4	3.9

K/A Statement:	Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation: -Trip of the MFW and necessary Condensate pumps

Reference Title:	Reference #:	Section:	Page:	Rev:
1C1.3 AOP1 Review	P8197L-008	II.F.2	14-15	4
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM - UNIT 1	1C1.3 AOP1	2.4.22, 2.4.25	11	4W

Facility Learning Objective:	P8197L-008 #5
Question Source:	New
Comments:	

Record #	**112**
RO #	**88**
SRO #	**87**

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- Containment pressure has risen to 28 psig
- All Containment Fan Cooler Units are operating
- Train B Containment Spray is operating
- One of the Containment Vacuum Breakers has been determined to be leaking past its seat at the rate of 0.5 psig per hour at 28 pisg

What would the effect over the next 4 hours be if NO Containment Spray were available?

Answers/Distracters:

a. Containment pressure would be HIGHER AND the total release from containment would be HIGHER.

b. Containment pressure would be HIGHER AND the total release from containment would be LOWER.

c. Containment pressure would be LOWER AND the total release from containment would be HIGHER.

d. Containment pressure would be LOWER AND the total release from containment would be LOWER.

Answer:	**LOK**	**Tier:**	**LOD**	**RO Group:**	**SRO Group:**	**Facility:**	**Exam Date:**
a	2-DR,DS	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- Use of a Containment Spray Pump will provide a more rapid pressure reduction in the containment. This reduction will reduce the leakage rate. If the leakage rate is reduced, over the 4-hour period the corresponding total release will also be smaller. The other choices provide coverage of the spectrum of choices.
b	Incorrect- If it is considered that the total release is not related to (or inversely related) to containment pressure.
c & d	Incorrect- This may be considered since accident analysis indicates the minimum requirement for protection of containment from overpressurization is 3 CFCUs operating. Although this is Adequate to protect containment, the pressure inside will decrease at a lower rate and the total leakage will be higher.

K/A System/Evolution:	**K/A #:**	**KAVRO**	**KAVSRO**
069 Loss of Containment Integrity	AK1.01	2.6	3.1

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to Loss of Containment Integrity: -Effect of pressure on leak rate

Reference Title:	**Reference #:**	**Section:**	**Page:**	**Rev:**
Containment System	P8180L-001	III.A.1	8-9	2
Containment Spray System	P8180L-002	I.C.2	9-10	4

Facility Learning Objective:	P8180L-001 #1 P8180L-002 #1,5
Question Source:	New
Comments:	

Record #	113
RO #	89
SRO #	88

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- The actions of 1E-0 "Reactor Trip Or Safety Injection" have been completed
- Entry in 1E-1 "Loss Of Reactor OR Secondary Coolant" was made
- 1FR-Z.1 "Response To High Containment Pressure" has been entered due to an ORANGE condition for the CONTAINMENT Critical Safety Function (CSF)
- Following completion of the actions of 1FR-Z.1, the ORANGE condition still exists
- NO other RED or ORANGE CSF conditions exist

What is the action that should be taken?

Answers/Distracters:

a. **Repeat** the sequence of steps of 1FR-Z.1 ONCE, THEN **return** to 1E-1.

b. **Return** to 1E-1 at the step in effect, AND 1FR-Z.1 does NOT need to be repeated again.

c. **Return** to 1E-1 at the step in effect AND **repeat** actions of FR-Z.1 in 10 minutes.

d. **Repeat** actions in 1FR-Z.1 UNTIL the ORANGE condition clears, OR a higher ORANGE or RED condition occurs.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-P	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-) These may be considered since this is a typical response for an ORANGE or RED condition. Some procedures (FR-C.1 direct the operator to return to initial steps if some actions were ineffective).
b	Correct- Step 7 of FR-Z.1 directs the operator to the procedure & step in effect. This is true even if the challenge is not removed and a transition back to FR-Z.1 should not be made since all possible actions have already been performed.
c	Incorrect- The ten minute time is taken from the procedural direction for reviewing the CSFSTs if the highest condition(s) is YELLOW. The crew is to perform tree scanning at a frequency of 10-20 minutes.
d	Incorrect-) These may be considered since this is a typical response for an ORANGE or RED condition. Some procedures (FR-C.1 direct the operator to return to initial steps if some actions were ineffective).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
069	Loss of Containment Integrity	AK3.01	3.8	4.2

K/A Statement:	Knowledge of the reasons for the following responses as they apply to the Loss of Containment Integrity: -Guidance contained in EOP for loss of containment integrity

Reference Title:	Reference #:	Section:	Page:	Rev:
F/FR Review	P8197L-014	VI.A.2.a	35-36	2
EOP Intro-Procedure Review	P8197L-010	III.E.4	21	2
Background Information for 1FR-Z.1 Response To High Containment Pressure	1FR-Z.1	Procedure Step 7	2	4

Facility Learning Objective:	P8197L-014 #28, 32 P8197L-010 #4,6
Question Source:	NRC Exam Bank
Comments:	DC Cook 1998 NRC Exams (RO/SRO) question [RO.AK19].

Record #	114
RO #	90
SRO #	89

Stem:

Given the following conditions on Unit 1:

- A reactor trip and safety injection have occurred due to a small RCS leak
- The operators are performing action of 1ES-0.1 "SI Termination" directing reset of SI
- Reactor trip breaker RTA is CLOSED AND reactor trip breaker RTB is OPEN
- RCS Tcold is 510°F AND decreasing
- RCS wide range pressure is 1750 psig AND steady
- Containment pressure is 3.6 psig AND slowly increasing

When the SI RESET buttons are depressed AND released, which of the following occurs?

Answers/Distracters:

a. The Train "A" SI actuation signal RESETS, THEN ACTUATES when the buttons are released.

b. The Train "A" SI actuation signal RESETS THEN ACTUATES when containment pressure exceeds 4 psig.

c. The Train "A" SI actuation signal WILL NOT RESET until reactor trip breaker RTA open signal is generated.

d. The Train "A" SI actuation signal WILL NOT RESET unless the Train "A" SI BLOCK switch is taken to BLOCK.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-DR,RI	1	2	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- With the pushbutton for Train A SI reset depressed, the SI signal to actuate components is reset. However without the P-4 signal for Train A (RTA open), the reset is not sealed-in. If an SI signal is present, as it is with RCS pressure below the SI setpoint of 1815 psig, then the actuation signal is regenerated when the pushbuttons are released.
b	Incorrect-- If all the signals in the stem of the question did not exceeded the SI actuation setpoints, then SI would be reset, and when containment pressure exceeded its SI actuation setpoint, 4 psig, then SI would reactuate.
c	Incorrect-- The SI signal is reset as long as the RESET pushbuttons (Train "A" specifically)are depressed, but would not block any actuation signal once the pushbutton is released (seal-in does not occur). Providing a (false) open signal for RTA would allow the seal-in to be completed and also block any further automatic SI actuation.
d	Incorrect- This is similar to C). The SI signal is NOT reset. However by taking the Pressurizer SI Unblock-Block Switches to BLOCK it will prevent the SI from occurring due to low Przr pressure or low steamline pressure. It will not prevent SI from occurring due to Containment Pressure.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E02	SI Termination	EK2.1	3.4	3.9

K/A Statement:	Knowledge of the interrelations between the SI Termination and the following: -Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference Title:	Reference #:	Section:	Page:	Rev:
Engineered Safeguards System	P8180L-006	V.B.1.d & e	20	3
Logic Diagrams Safeguards Actuation Signals	X-HIAW-1-242	SI Reset & Block		D

Facility Learning Objective:	P8180L-006 #4.a
Question Source:	Facility Exam Bank
Comments:	Requal Bank question P8180L-006 004. Changed some values in premise. Changed selection "C" to prevent diametric opposed to correct answer. Changed to include possible jumper/override for P-4 generation.

Record #	115
RO #	
SRO #	90

Stem:

Given the following conditions on Unit 1:

- A small break LOCA has occurred
- 1ES-1.1 "Post LOCA Cooldown And Depressurization" is being performed
- ADVERSE Containment conditions do NOT exist
- Subcooling has been maintained ABOVE 25°F while in 1 ES-1.1
- Pressurizer level has been maintained ABOVE 28% while in 1 ES-1.1
- Level in the RWST has been maintained ABOVE 50% while in 1 ES-1.1

Which of the following describes the expected status of the plant when the actions of 1ES-1.1 are complete (Tavg at 200°F)?

Answers/Distracters:

a.
- ONE SI Pump RUNNING
- SI accumulators UNISOLATED
- One train of RHR ALIGNED for recirculation

b.
- BOTH SI Pumps STOPPED
- SI accumulators UNISOLATED
- ONE train of RHR ALIGNED for shutdown cooling

c.
- ONE SI Pump RUNNING
- SI Accumulators ISOLATED
- ONE train of RHR ALIGNED for shutdown cooling
- ONE train of RHR ALIGNED for injection

d.
- BOTH SI Pumps STOPPED
- SI Accumulators ISOLATED
- ONE train of RHR ALIGNED for injection
- ONE train of RHR ALIGNED for shutdown cooling

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
d	3-SPK	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-- SI Pumps would be stopped, Accumulators isolated and RHR is not aligned for recirc until RWST level falls below 33%. Could be considered as E-1 may result in such conditions (if RWST Lo Level).
b	Incorrect-Accumulators would be isolated and the second train of RHR would be operating in injection mode. This could be considered since the SI pump could be used for inventory maintenance and the RHR Pump used for cooling
c	Incorrect-- SI Pump would not be running in this situation. This could be considered if either Przr level or subcooling could not be maintained.
d	Correct- The direction of ES-1.1 has a result that at 200°F, both RHR systems will be operating, one aligned for shutdown cooling and one aligned for injection flow to the RCS. Both SI Pumps are stopped with the given conditions; the second on stopped when RCS temperature is less than 335°F. Accumulators are isolated when the RCS is cooled down and depressurized.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
E03 LOCA Cooldown and Depressurization	EA2.2	3.5	4.1

K/A Statement:	
	Ability to determine and interpret the following as they apply to the LOCA Cooldown and Depressurization: -Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-1/E-2 Review	P8197L-012	X.A.5 X.C.6	55, 57	2
Post LOCA Cooldown And Depressurization	1ES-1.1	Steps-12,18,25	9, 12, 16	15

Facility Learning Objective:	P8197L-012 #2,3
Question Source:	New
Comments:	

Record #	116
RO #	
SRO #	91

Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred
- The Emergency Director has declared an ALERT
- The crew is performing the actions of 1ES-1.1 "Post LOCA Cooldown And Depressurization"
- Annunciators **47016-0204** 11 RWST LO LVL and **47019-0503** 11 RWST LO LVL have just alarmed

What action must be taken?

Answers/Distracters:

a. A Site Evacuation must be **ordered** AND the people **directed** to assemble at the PI Training Center.

b. A plant **announcement** must be made warning personnel to restrict entry into the Auxiliary Building due to potential high radiation.

c. The event must be **reclassified** as a GENERAL EMERGENCY AND the NRC, State and local governments **notified** within 15 minutes.

d. Protective Action Guidelines (PAGs) **determined**, AND the State and local governments **notified** within 15 minutes following evaluation.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	2-DR	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- A Site Evacuation is only required if the event has been classified as an SITE AREA emergency. Also this action is incorrect as to designates location. The PITC is designated only if both of the other designated assembly points are most likely uninhabitable.
b	Correct-The condition given will have the crew enter 1ES-1.2 to initiate a transfer to recirculation phase. In accordance with 1ES-1.2, the initial CAUTION warns that the actions may cause high radiation in the Aux. Bldg. The Background Information indicates any personnel in those areas must be alerted to this danger
c	Incorrect-- There is no reason for the given information to upgrade the event to a GENERAL EMERGENCY. The NRC should be notified ASAP of any escalation of Emergency event, and the state/local government within 15 minutes.
d	Incorrect- PAGs should be the states (State Governors and Departments of Health) ASAP. The states may direct as a contingency action that the plant may recommend immediate protective actions to County officials for a fast-developing incident. The given conditions do not indicate any PAGs need be addressed.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
E03 LOCA Cooldown and Depressurization	2.1.14	2.5	3.3

K/A Statement:	Conduct Of Operations -Knowledge of system status criteria which require the notification of plant personnel.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-1/E-2 Review	P8197L-012	XI.D	60	2
Background Information For 1ES.1.2 TRANSFER TO RECIRCULATION	1ES-1.2	Basis for Actions Caution, Step 1	1	14

Facility Learning Objective:	P8197L-012 #6
Question Source:	New
Comments:	

Record #	117
RO #	91
SRO #	92

Stem:

Which of the following events results in a LOCA in the Auxiliary Building if NO operator action is taken?

Answers/Distracters:

a. **MV-32202**, SI Test Line to RWST FAILS to CLOSE during recirculation following a LOCA, AND a RUPTURE of the 11 RWST.

b. 11 RCP Thermal Barrier Heat Exchanger tube RUPTURE coincident with a LOSS of instrument air to Unit 1 Containment AND a PIPING BREAK at the CCW Surge Tank outlet.

c. A steam generator TUBE RUPTURE in 12 SG coincident with a LOCKOUT of Bus 16 with steam generator blowdown in service, AND a RUPTURE of 11SGB Flash Tank.

d. FAILURE of "Train A" Containment Isolation during a small-break LOCA with Excess Letdown in service, AND a PIPING BREAK at the Seal Water Heat Exchanger outlet.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	3-SPK	1	4	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-The SI Test Line is isolated by two normally open MVs. MV-3220 and MV32203. These valves are interlocked with the Containment Sump B Isolation valves (MV-32075, 32076, 32077 and 32078) such that either MV-32203 OR MV-33202 must be closed to open the valves. With MV-32203 closed, no loss of recirculating coolant from the SI Pump discharge can be expected (Return to RWST isolated).
b	Correct- In the event of failure in the Thermal Barrier HX high flow is sensed in the Component Cooling return line and CV-31245, , will close in the AUTO position. This is an air-operated valve inside containment and fails open upon the loss of instrument air. (The operator may take manual action to isolate the line coming out of containment by taking MV-32090.)
c	Incorrect-Excess letdown use the RCP seal return line as the return flowpath out of containment. This line uses two containment isolation valves, MV-32199 and MV-32166 to isolate the containment penetration. MV-32199 receives a Train B Containment Isolation (CI) to close while MV-32166 receives a Train A CI signal to close. MV-32166 will remain open but MV-32199 will close isolating the excess letdown flow to the seal water HX
d	Incorrect- The steam generator blowdown uses two containment isolation valves in the blowdown line from each SG to isolate SGB during accident conditions. One of set of valves (MV-32043) is powered from Train A electrical and closes on Train A CI signal. The other valve (MV-32058) is powered from Train B electrical and closes on Train B CI signal. Therefore this valve would remain open while MV-32043 closes, isolating SG 12 blowdown to the Flash Tank.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E04	LOCA Outside Containment	EA1.1	4.0	4.0

K/A Statement:	
	Ability to operate and/or monitor the following as they apply to the LOCA Outside Containment: -Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference Title:	Reference #:	Section:	Page:	Rev:
Component Cooling	P8172L-002	IV.A.4.g IV.B.10	16,20	4
Interlock Logic Diagram Component Cooling System	NF-40321-2			U
Component Cooling System Flow Diagram	NF-39245-1	F-5		L

Facility Learning Objective:	P8172L-002 #2.d, 5.f
Question Source:	new
Comments:	

Record #	118
RO #	92
SRO #	93

Stem:

Given the following conditions on Unit 1:

- Safety Injection has actuated
- A LOCA has been identified at the flange for flow transmitter 1FE-626 (inputs control for positioning RHR Heat Exchanger Bypass valve) on the RHR return line to loop B RCS

Which of the following must be performed to ISOLATE the leak, while MINIMIZING the affect on the ECCS system operation?

Answers/Distracters:

a. - **Stop** 11 RHR Pump
   - **Close** valve **MV-32065**, RHR TO RX VSL

b. - **Stop** 12 RHR Pump
   - **Verify MV-32066**, RHR TO RC LOOP B COLD LEG is closed
   - **Close MV-32065**, RHR TO RX VSL

c. - **Stop** BOTH of the RHR Pumps
   - **Verify MV-32066**, RHR TO RC LOOP B COLD LEG is closed
   - **Close MV-32065**, RHR TO RX VSL.

d. - **Verify** valve **MV-32066**, RHR TO RC LOOP B COLD LEG is closed
   - **Open RH-2-5** AND **RH-2-6**, 11/12 RHR HX CROSSOVER, OUTLET valves.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	3-SPK	1	4	2	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-As stated in (b) the RHR Train "A" header is separate from the leak. Stopping the 11 RHR Pump would not affect the leak, and would decrease delivery of ECCS flow to RCS (if RCS pressure allows injection). Also MV-32065 is associated with the RHR Train "B" low head SI line, not Train "A" (MV-32064). The 12 RHR Pump and the path from RCS via MV-32066 is available to leakage location
b	Correct-The LOCA is located on the return header for RHR to loop B outside containment. Additionally this header serves a part of the line for RHR Pump B discharge to the reactor vessel (SIS low head injection) where the line branches inside containment upstream of the 2 isolation valves: MV-32066, RHR TO RC LOOP B COLD LEG (normal RHR return) and MV-32065, RHR TO RX VSL (SIS Injection). ECA-1.2 specifically directs closure of MV-32066 to attempt to isolate the leak. During normal at-power alignment, the "A" and "B" Train RHR discharge headers are separate by closure of the two crossover isolation valves. RHR Pump A then delivers flow through a separate header which is unaffected by the leak. So, to stop the leak and minimize impact on ECCS the 12 RHR Pump must be stopped (flowpath would not be available and also forced flow feeds leak) and the RHR "B" Train header isolated from the RCS by closing the two MVs mentioned above
c	Incorrect- It is not necessary to stop both pumps since RHR Train A does not supply the leak in current configuration. This impacts the ability to inject to RCS from RHR and operation of ECCS during recirculation phase. Therefore this option does not minimize affect on ECCS.
d	Incorrect- Closing MV-32066 is correct as states in ECA-1.2; however, opening the crossover isolation valves will result in both RHR pumps feeding the leak. Also not closing MV-32065 fails to isolate that potential leak path.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E04	LOCA Outside Containment	EK2.2	3.8	4.0

K/A Statement:	Knowledge of the interrelations between the LOCA Outside Containment and the following: -Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-1/E-2 Review	P8197L-012	XIV.C.1	70	2
LOCA OUTSIDE CONTAINMENT	1ECA-1.2	1.b	3	3
Flow Diagram Residual Heat Removal	X-HIAW-1-31			M

Facility Learning Objective:	P8197L-012 #10
Question Source:	New
Comments:	

Record #	119
RO #	93
SRO #	94

Stem:

Given the following conditions on Unit 1:

- The operators are performing FR-H.1, "Response to Loss of Secondary Heat Sink"
- RCS Tavg is 530°F
- Pressurizer pressure is 1900 psig
- SI has been reset and the initiating condition cleared
- Reactor trip breakers are open

Which of the following describes ALL the actions, in CORRECT ORDER, required to feed the SGs using the Feedwater Bypass valves?

Answers/Distracters:

a.
- **Start** ONE main FW pump
- **Depress** BOTH sets of FW BYPASS RESET pushbuttons
- **Throttle** OPEN the FW Bypass valves in MAN

b.
- **Reset** Containment Isolation
- **Open** FW containment isolation valves
- **Start** ONE main FW pump
- **Throttle** OPEN the FW Bypass valves in MAN

c.
- **Reset** Containment Isolation
- **Open** FW containment isolation valves
- **Depress** BOTH sets of FW BYPASS RESET pushbuttons
- **Start** ONE main FW pump
- **Throttle** OPEN the FW Bypass valves in MAN

d.
- **Cycle** the reactor trip breakers
- **Open** FW containment isolation valves
- **Start** ONE main FW pump
- **Depress** BOTH sets of FW BYPASS RESET pushbuttons
- **Throttle** OPEN the FW Bypass valves in MAN

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
c	1-P	1	3	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- If a Containment Isolation signal is not determined to have been generated, then the FW Isolation valves would remain open. The FW pump trip signal from SI also has a logic path that trips the FW Pumps separate from the seal-in circuit that goes through the FW RESET circuit. This circuit, if considered alone, does not prevent FW pump start without FW RESET.
b	Incorrect– This implements the consideration for the separate FW RESET circuit affecting the FW pumps (as described above), The FW pumps should have tripped on the SI and do not restart on clearing the SI until FW reset circuitry has been actuated.
c	Correct-With the SI, the main FW Pumps are tripped, the FW Isolation valves (MV-32023 & MV-32024) close on the Containment Isolation signal, and both the FW Main valves and FW Bypass valves are closed (SI). To establish a feed path, a feed pump must be started, the FW Isolation valves must be opened and the FWQ Bypass valve must be opened. The flow path is established by resetting the CI signal in order to open the FW Isolation valves. The FW BYPASS RESET pushbuttons are pushed to remove the SI seal-in signal that trips the main FW pumps, closes the FW Bypass valves (and trips the main turbine). So the FW BYPASS RESET pushbuttons are depressed, one FW pump started and the FW Bypass valve throttled open to establish flow
d	Incorrect- This is related to the normal operation of the FW system as related to a non-accident condition (reactor trip). The FW Main valves cannot be opened with Low Tavg and the reactor tripped, due to another seal-in circuit. This also treats the Main FW pump as in A).

K/A System/Evolution:		*K/A #:*	*KAVRO*	*KAVSRO*
E05	Loss of Secondary Heat Sink	EK2.1	3.7	3.9

K/A Statement:	Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: -Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
Steam Generator Level Control System	P8174L-006	IV.D.2.d	28-29	3
Feedwater Control & Isolation	X-HIAW-1-244			D
Interlock Logic Diagram Feedwater System	NF-40310-1&2			N&L
RESPONSE TO LOSS OF SECONDARY HEAT SINK	1FR-H.1	Step 4.c & d	5	11

Facility Learning Objective:	P8174L-006 #1,6
Question Source:	New
Comments:	

Record #	120
RO #	94
SRO #	95

Stem:

Given the following conditions on Unit 1:

- The operators are evaluating the Critical Safety Function Tree F-0.2, "Core Cooling"
- One RCP is currently running
- The operators are evaluating RVLIS Dynamic Head reading against the Table listing of 32%

What is the basis for verifying this RVLIS value?

Answers/Distracters:

a. A RVLIS reading GREATER THAN 32% means RVLIS Upper Head Range will be on scale if the RCP is stopped.

b. A RVLIS reading GREATER THAN 32% means core inventory has recovered to the point that SI accumulators may be isolated.

c. A RVLIS reading LESS THAN 32% means actual RCS voiding is greater than 50% and if the RCP is stopped, the core may not remain covered or adequately cooled.

d. A RVLIS reading LESS THAN 32% means RCS voiding will cause RCP cavitation, requiring stopping of the RCP in FR-C.2, "Response to Degraded Core Cooling".

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	1-P	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- The relationship between Dynamic and Upper Head range is not of importance. Each specific range is used to evaluate core condition (coverage) with differing flow condition. Also Upper Head Range is not used for Core Cooling evaluation but, the Full Range is. RVLIS full range indicates collapsed liquid level never below top of core indicates core damage should not have occurred.
b	Incorrect- This is not directly evaluated by the value, but is used in 1FR-C.2 not to evaluate if the Accumulator should be isolated, but if the Accumulator Isol valves should be verified open (step following check of RVLIS < 32%) to allow injection of Accumulator inventory as RCS is cooled and depressurized. RCS hot leg temperature is used to evaluate Accumulator isolation.
c	Correct- The specific branch for RVLIS evaluation with RCP(s) running checks for RCS voiding less than 50%, which, if the RCP(s) is subsequently stopped, would ensure the core would initially be kept covered and adequately cooled. If RVLIS dynamic range head is less than 32% with one RCP running (< 62% with both RCPs running), then a degraded core cooling condition exists.
d	Incorrect- The value is not specific for potential RCP cavitation. An RCP should be run since single-phase or two-phase forced flow provides better core cooling than natural circulation flow. While cavitation is a normal consideration for RCP operation, it is not for loss of/degraded core cooling scenarios. RCPs are only stopped 1) if adequate cooling has been established (RCS temp) or in FR-C.1 prior to SGs being depressurized to atmospheric due to anticipated loss of #1 seal requirements (prevent damage/leakage due to seal damage).

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E07	Saturated Core Cooling	EK2.1	3.2	3.5

K/A Statement:	Knowledge of the interrelations between the Saturated Core Cooling and the following: -Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference Title:	Reference #:	Section:	Page:	Rev:
F/FR Review	P8197L-014	III.A.2.f	16	2
Background Information for1F-0.2 Core Cooling Status Tree	1F-0.2	Basis - Branch Description: RVLIS Dynamic Head	3	4

Facility Learning Objective:	P8197L-014 #9
Question Source:	Facility Exam Bank
Comments:	Initial Bank question P8197L-014 008. Change to premise to direct to specific condition and value. No changes to selections.

Record #	121
RO #	95
SRO #	

Stem:

Given the following conditions on Unit 1:

- A natural circulation cooldown is in progress per 1ES-0.3A "Natural Circulation Cooldown With CRDM Fans"
- Pressurizer pressure is being reduced by cycling **CV-31329**, AUX PRZR SPRAY FROM REGEN HX
- Charging and letdown were in manual and balanced during the cooldown
- Pressurizer level fell from 19% to 14% during the cooldown
- As pressure is being lowered through 1300 psig, a rapid increase is noted in Pressurizer level to 32%

What action should be taken by the operator?

Answers/Distracters:

a. **Close** BOTH SG PORVs.

b. **Energize** AND **close** BOTH SI Accumulator Isolation valves.

c. **Place** EACH Pressurizer Heater Group control switch to **OFF** AND THEN to **ON.**

d. **Place** EACH Pressurizer Heater Group control switch to **OFF** AND **open CV-31329**, AUX PRZR SPRAY FROM REGEN HX.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	3-SPK	1	4	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-This will stop any cooldown in progress and will result in RCS heatup. This is considered if the cooldown is responsible (which action would be taken in response to a loss of Przr level). This action in this case will only reduce subcooling and exacerbate the voiding problem
b	Incorrect- The SI accumulators are normally isolated during cooldown and depressurization. If they are not they will inject coolant into the RCS resulting in a mass increase in RCS, and higher indicated Przr level. However, in this case it does not apply, since the RCS pressure is well above the maximum Accumulator pressure of 770 psig, preventing injection. The isolation is normally directed to be performed at about 1000 psig.
c	Correct- During natural circulation cooldown with charging and letdown balanced a rapid increase in Przr level during depressurization results from the voiding occurring in the reactor head. The coolant in the vessel head is displaced into the Przr. Energizing the Przr heaters will heat the coolant in the Przr, raising the pressure. As the pressure rises it will result in decrease voiding in the vessel head. The heaters are required to placed to OFF initially (to reset the interlock) since Przr level had fallen below the low-low setpoint - isolating letdown and securing heaters.
d	Incorrect- This is the inverse of the correct action resulting in further depressurization of Przr steam space and more voiding in the vessel/vessel head.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E09	Natural Circulation Operations	EA1.1	3.5	3.5

K/A Statement:	Ability to operate and/or monitor the following as they apply to the Natural Circulation Operations: -Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	VI.A VI.F.2.d	16-17 21	2
Pressurizer Level Control System	P8170L-006	IV.B.4	17	3
Alarm Response Procedures	C47012-0607	Subsequent Action 2	1	32

Facility Learning Objective:	P8197L-011 #16,17 P8170L-006 #7
Question Source:	Significantly modified from NRC Exam Bank
Comments:	Byron 9/1998 NRC Exam. Changed the premise to include the letdown/heater interlock response. Changed all selections to specific actions that result in expected conditions (as given in Byron question).

Record #	122
RO #	96
SRO #	96

Stem:

Given the following conditions on Unit 1:

- A natural circulation cooldown is in progress per 1ES-0.4 "Natural Circulation Cooldown With Steam Void In Vessel"
- RCS pressure is 1200 psig
- Pressurizer level is 32%
- RVLIS Full Range reads 88%
- 12 RCP seal DP reads 325 psid
- 12 RCP #1 seal leakoff flow reads 0.8 gpm
- 12 RCP is reported to now be available for starting (power restored to bus)

Which of the following must be completed prior to attempting to start 12 RCP?

Answers/Distracters:

a. **Raise** Pressurizer level to GREATER THAN 84%.

b. **Raise** RCS pressure to GREATER THAN 1250 psig.

c. **Raise** seal injection flows to INCREASE seal DP to GREATER THAN 400 psid.

d. **Raise** #1seal leakoff flow to GREATER THAN 1 gpm by **opening** the No. 1 Seal Bypass Isolation Valve.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	2-RI	1	3	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Correct- With RVLIS full range indicating less than full (< 93%), additional restrictions are placed on Przr level and RCS subcooling to be met prior to starting an RCP in order to accommodate any void collapse. Przr level is low such that the collapse of the void upon RCP start would result in loss of all Przr level. The RCP #1 seal dP and seal leakoff flow are within the normal limits included in the starting a RCP, as detailed in C3
b	Incorrect- RCS pressure is not a limitation for starting the RCP in "normal" (non-accident) situations. 1250 psig is used because in the event of a LOCA 9or secondary break), the value of 1250 psig is the pressure limit below which the RCP should be stopped (prior to any controlled cooldown, for small break LOCA condition)
c	Incorrect-Seal dP is one of the items considered for starting an RCP. Under normal conditions the dP is expected to be > 400 psig; however, the limits for consideration in starting an RCP > 200 psid (with operating limits between 200 and 2470 psid). Control Room Instrumentation for the RCP dP has a maximum indicated value of 400 psid
d	Incorrect-Seal leakoff flow is low but within the allowed limits 0.25 gpm and 5.0 gpm. It is also above the value below which the operator is directed to the AOP for #1 seal problems. Also it is not allowed to open the Seal Bypass valves with RCS pressure > 1000 psig

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
E10 Natural Circulation with Steam Void in Vessel with/without RVLIS	EA1.3	3.4	3.7

K/A Statement:	Ability to operate and/or monitor the following as they apply to the Natural Circulation with Steam Void in Vessel with/without RVLIS: -Desired operating results during abnormal and emergency situations.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-O Review	P8197L-011	VI.F.1.c	21	2
Reactor Coolant Pumps	P8170L-002	IV.A.7.h.6	17	2
Natural Circulation Cooldown With Steam Void In Vessel	1ES-0.4	Step 1.b	3	8

Facility Learning Objective:	P8197L-011 #17 P8170L-002 #11
Question Source:	New
Comments:	

Record #	123
RO #	97
SRO #	97

Stem:

Given the following conditions on Unit 1:

- The reactor tripped from 100% power
- One Code Safety Valve on 11 SG OPENED AND failed to reclose
- SI was manually actuated
- 12 AFW pump could NOT be started
- 12 SG experienced a pipe shear inside containment
- The crew is performing the actions of 1ECA-2.1 "Uncontrolled Depressurization of Both Steam Generators"
- SG pressures are currently: 600 psig for 11 SG and 450 psig for 12 SG

What action should be taken concerning feed to the SGs?

Answers/Distracters:

a. **Isolate** all feed and steam paths to Both SGs THEN **Align** the Unit 1 Condensate system to provide feed to the SGs.

b. **Isolate** all feed and steam paths to Both SGs THEN **enter** FR-H.1 "RESPONSE TO LOSS OF SECONDARY HEAT SINK".

c. **Maintain** the steam supply to 11 AFW Pump from 11 SG OPEN AND **feed** SGs using 11 AFW pump until a different source of feedwater can be aligned.

d. **Open** both Pressurizer PORVs to provide a feed and bleed path THEN **isolate** all feed and steam paths to BOTH SGs.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
c	2-RI	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect- This action may be considered since the condensate system is a source capable of providing water to the SGs but isolating the steam supply to 11 Aux. feedwater pump would result in a degraded system condition and result in a transition to FR-H.1. Therefore this isolation must not be performed.
b	Incorrect-Although FR-H.1 would restore feedwater sources to the SGs it is inappropriate to intentionally isolate all sources of feedwater and then enter a FR procedure.
c	Correct-If the turbine driven AFW pump is the only available source of feed flow to the SGs then isolation of its steam flow path would, as mentioned in a above, degrade system conditions and must not be performed. When a different source of water becomes available then isolation would be appropriate.
d	Incorrect-Aligning a feed and bleed path for core cooling is only appropriate when no heat sink is available. With 11 AFW pump running and water in the SGs, a heat sink is still available and feed and bleed should not be performed.

K/A System/Evolution:		K/A #:	KAVRO	KAVSRO
E12	Uncontrolled Depressurization of all Steam Generators	EK1.2	3.5	3.8

K/A Statement:	Knowledge of the operational implications of the following concepts as they apply to the Uncontrolled Depressurization of all Steam Generators: -Normal, abnormal and emergency operating procedures associated with Uncontrolled Depressurization of all Steam Generators.

Reference Title:	Reference #:	Section:	Page:	Rev:
E-1/E-2 Review	P8197L-012	V.E	37	2
Uncontrolled Depressurization of Both Steam	1ECA-2.1	CAUTION prior to step 1	3	12

Facility Learning Objective:	P8197L-012 #6
Question Source:	new
Comments:	

Record #	**124**
RO #	**98**
SRO #	**98**

Stem:

Given the following conditions on Unit 1:

- A steam line break has occurred on the "B" loop main steam header downstream of 12 MSIV
- A common mode failure has PREVENTED CLOSURE of both MSIVs
- RCS cooldown rate is 160°F/hr
- Both SG WR levels are less than 50%

When the appropriate actions are taken, what final AFW flow will be established?

Answers/Distracters:

a. 200 gpm to each SG.

b. 200 gpm to 11 SG and 0 gpm to 12 SG.

c. 160 gpm to 11 SG and 40 gpm to 12 SG.

d. 40 gpm to each S/G.

Answer:	*LOK*	*Tier:*	*LOD*	*RO Group:*	*SRO Group:*	*Facility:*	*Exam Date:*
d	1-P	1	2	1	1	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-In an accident condition, with SG levels less than 5% NR, total feed flow is to be maintained greater than 200 gpm until at least one SG level is > 5% NR. It is typical to maintain 200 gpm to each SG
b	Incorrect-As stated 200 gpm is the minimum flow for conditions other than depressurization of both SGs. If the least-affected SG is selected it may be considered to receive the full required flow while flow to the most-affected SG is stopped. This is not in agreement with the CAUTION prior to the step which requires the 40 gpm flow to each SG
c	Incorrect- This combines the required 40 gpm flow to a SG with the minimum 200 gpm flow total.
d	Correct-With the cooldown rate greater than 100°F/hr during uncontrolled depressurization of both SGs, feed flow is reduced to 40 gpm to each SG if NR level is < 5%.

K/A System/Evolution:	*K/A #:*	*KAVRO*	*KAVSRO*
E12 Uncontrolled Depressurization of all Steam Generators	EK2.2	3.6	3.9

K/A Statement:	Knowledge of the interrelations between the Uncontrolled Depressurization of all Steam Generators and the following Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Reference Title:	*Reference #:*	*Section:*	*Page:*	*Rev:*
E-1/E-2 Review	P8197L-012	V.D.2	35	2
Uncontrolled Depressurization Of Both Steam Generators	1ECA-2.1	Step 2	3	12

Facility Learning Objective:	P8197L-012 #15
Question Source:	Facility Exam Bank
Comments:	Requal Part B Bank question P8197L-012 011. Changed layout of premise only

Record #	125
RO #	99
SRO #	99

Stem:

Which of the following is correct concerning an Orange Path (Containment Sump "B" level greater than 8ft) in the Containment Critical Safety Function Status Tree?

Answers/Distracters:

a. Continued Core cooling CANNOT be assured since the entire contents of the RWST has been injected into Containment.

b. Critical plant components needed for plant recovery could be damaged and rendered inoperable due to flooding in Containment.

c. Auxiliary Feedwater to a SG faulted in Containment must remain isolated even if required for cooldown of the RCS.

d. Cooling Water to CFCUs must NOT be isolated since CFCUs are required to maintain Containment pressure less than 46 psig.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
b	1-F,P	1	3	3	3	Prairie Island	9/10/2001

Basis for answers:

a	Incorrect-Even though the entire contents of the RWST has been injected into Containment, core cooling is established by transferring the RHR pumps to the recirc mode.
b	Correct-The maximum level of water in the Containment sump following a major accident is based on the entire water contents of the RCS, RWST, CST, and the SI Accumulators. This water level approximates the maximum water volume introduced into Containment following a steam or feedline break inside Containment followed by feed and bleed cooling of the core via the SI pumps and PRZR PORVs. The critical systems and components necessary to ensure an orderly safe shutdown of the plant and to provide feedback to the operators concerning the conditions of the core and RCS are generally located above this water level.
c	Incorrect-The addition of Aux. feedwater to a faulted SG would add water to sump "B" which is undesirable at this point, but maintaining core cooling is of the utmost importance and should not be terminated due to high containment water level.
d	Incorrect-Cooling Water could be a major source of water leakage into containment. CFCUs aid in limiting the containment pressure rise but are not required since the Containment Spray system is sized such that containment pressure is limited to less than design pressure.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
E15 Containment Flooding	EK2.2	2.7	2.9

K/A Statement: Knowledge of the interrelations between the Containment Flooding and the following:
-Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Reference Title:	Reference #:	Section:	Page:	Rev:
FR Procedure review	P8197L-014	VI.A.2.b	37	2
Response To High Sump B Level	1FR-Z.2	Summary	1	3

Facility Learning Objective:	P8197L-014 #28
Question Source:	New
Comments:	

Record #	126
RO #	100
SRO #	100

Stem:

A LOCA in Containment is in progress and the Reactor operator has recorded Containment parameters as follows:

Time	Cont. Radiation	Cont. Pressure
0812	8.1E3 R/hr	3.8 psig
0815	1.2E4 R/hr	4.5 psig
0831	1.4E5 R/hr	4.9 psig
0838	5.0E5 R/hr	5.8 psig

When was the first time Adverse Containment parameters were required to be used, and for how long will use of the Adverse Containment numbers be in effect?

Answers/Distracters:

a. Adverse Containment was first entered at 0815, AND is in effect for the entire time the crew is in the EOPs.

b. Adverse Containment was first entered at 0815, AND is in effect until containment parameters drop below the Adverse Containment setpoints.

c. Adverse Containment was first entered at 0838, AND is in effect for the entire time the crew is in the EOPs.

d. Adverse Containment was first entered at 0838, AND is in effect until containment parameters drop below the Adverse Containment setpoints.

Answer:	LOK	Tier:	LOD	RO Group:	SRO Group:	Facility:	Exam Date:
a	1-P	1	2	2	2	Prairie Island	9/10/2001

Basis for answers:

a	Correct-Adverse containment conditions are defined as either containment pressure greater than 5 psig or containment radiation levels greater than 1E4 R/hr. At 0815 containment rad levels were noted to be 1.2E4 therefore adverse conditions were entered at that time. If containment rad levels go above the adverse setpoints the crew is required to use adverse containment numbers from that point on regardless if containment levels decrease below the adverse setpoints or not.
b	Incorrect-As stated above, adverse numbers are required to be used the entire time in the EOPs. This answer could be considered if the adverse rad levels aren't realized since the adverse pressure setpoint is only required to be used when pressure is above the setpoint as long as rad levels didn't go above the adverse Cont rad level setpoint.
c	Incorrect-This answer could be considered if the Cont. rad level adverse setpoint isn't realized.
d	Incorrect-As in b&c above, this answer could also be considered if rad levels and the requirement to stay in adverse numbers aren't used.

K/A System/Evolution:	K/A #:	KAVRO	KAVSRO
E16 High Containment Radiation	EA2.1	2.9	3.3

K/A Statement: Ability to determine and interpret the following as they apply to the High Containment Radiation:
-Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Reference Title:	Reference #:	Section:	Page:	Rev:
EOP intro-Procedure Review	P8197L-010	III.D.13.a&c	17,18	2
Reactor trip or Safety Injection	1E-0	Info page		4

Facility Learning Objective:	P8197L-010 #3.m
Question Source:	Facility exam bank question P8197L-011 084
Comments:	Modified to reflect correct answer and editorial changes.