PROPOSED 2006
BYRON WRITTEN INITIAL EXAMINATION,

QUESTIONS 1 THROUGH 75 RO EXAMINATION
QUESTIONS 76 THROUGH 100 SRO EXAMINATION

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
1	Both	1	1	000007	EK1.04	3.6	3.9	High

System/Evolution Name:
Reactor Trip

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the reactor trip:

KA Statement:
Decrease in reactor power following reactor trip (prompt drop and subsequent decay)

UserID:

Topic

Question Stem:

Unit 1 was operating at 100% power.
At Time=0 an inadvertent turbine trip occurs.
All systems respond as designed.

At Time=2 minutes, IR Nuclear Instrument N35 reads 5E-2% power.

Which of the following is the MINIMUM additional time expected for the POWER ABOVE PERMISSIVE P6 lite to GO OUT on the IR NI N35 Drawer on 1PM07J?

A 6 minutes

B 12 minutes

C 15 minutes

D 18 minutes

Answer: B

Task No:

Obj No: S.NI2-08-B, A.NT7-

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-NT-XL-07, Nuclear Theory Chapter 7, Neutron Kinetics, pg 33, 46, 60 Reference:
ILT Systems: I1-NI-XL-01BY, Gamma-Matric Source and Intermediate Range Nuclear Instrumentation, pg 11, 21 22, 28.

Explanation:
A prompt drop and rapid power level decay occurs over the first 2-4 minutes following a reactor trip. The reactor then reaches a stable -1/3 DPM startup rate due to the dominant decay of the long-lived fission product precursers. This stable period is sustained well into the source range. From the 5E-2% power to 5E-6% (Setpoint for the POWER ABOVE PERMISSIVE P6 lite) is 4 decades and will take ~12 minutes. MJJ 2/27/06

Date Written: 2/27/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
2	Both	1	1	000008	82.1.30	3.9	3.4	High

System/Evolution Name:
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Category Statement:
Conduct of Operations

KA Statement:
Ability to locate and operate components, including local controls.

UserID: Topic

Question Stem:

Unit-1 is at 100% power.
PZR PORV DSCH TEMP HIGH (1-12-C6) is in alarm.
PORV 1RY456 indicates partially OPEN after C/S was placed in CLOSE
1RY8000B, PORV Isolation Valve, C/S was taken to CLOSE but continues to show dual indication.
Charging Header flow has risen ~ 5gpm and stablized PZR level on program.
PZR Pressure is stable at 2218 psig
The crew has entered 1BOA PRI-1, Excessive Primary Plant Leakage.

What is/are the NEXT required action(s) based on these indications?

A Trip the Reactor and manually actuate Safety Injection.

B Locally CLOSE 1RY8000B at MCC 131X2 by resetting the breaker and re-CLOSING the breaker.

C Dispatch an operator with RP to enter Containment and manually CLOSE 1RY8000B.

D Locally CLOSE 1RY8000B at MCC 132X2 after placing the LOCAL/REMOTE switch at the breaker in LOCAL.

Answer: D

Task No:

Obj No: T.OA12-02

Time:

Cross Ref:

Question Source: New

Question Difficulty: Medium

Reference:
1BOA PRI-1, Excessive Primary Plant Leakage, Step 3 RNO
I1-OA-XL-12, 1BOA PRI-1, Excessive Primary Plant Leakage

Explanation:
A. is incorrect because the system has stablized within normal charging capacity and does not warrant a Reactor trip or SI.
B. is incorrect because this is the wrong MCC and there is no indication that the breaker has tripped requiring a reset.
C. is incorrect because this is not a procedural option nor would this occur at 100% power.
D is the correct action as specified for this condition in 1BOA PRI-1.

Date Written: 2/28/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
3	Both	1	1	000009	EA2.24	2.6	2.9	High

System/Evolution Name:
Small Break LOCA

Category Statement:
Ability to determine and interpret the following as they apply to a small break LOCA:

KA Statement:
RCP temperature setpoints

UserID: Topic

Question Stem:

A Small Break LOCA has just occurred on Unit 1 and Safety Injection has been actuated. All systems responded as designed.

Without any additional operator action, which of the following RCP temperatures would RISE significantly during the next 10 minutes of RCP operation?

A Thrust Bearing

B Motor Lower Radial Bearing

C Motor Stator Winding

D Pump Lower Radial Bearing

Answer: C

Task No:

Obj No: S.RC2-08-C

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RC-XL-02, Ch 13, Reactor Coolant Pump

Reference:

Explanation:

C. is correct because the SI and Phase A actuations cause RCFCs to run in Low Speed without Chilled Water for cooling (SX only) and Post-LOCA environment will cause higher temperatures in Cnmt. The Motor Windings are air cooled from the Cnmt atmosphere.
A and B are incorrect because CCW flow actually increases post SI (2 CC pumps running) and adds additional cooling for the oil coolers causing temperature to be stable or drop.
D. is incorrect since the additional seal injection flow rate post SI aids in cooling and temperature will be stable or drop.

Date Written: 3/2/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
4	Both	1	1	000022	AA2.04	2.9	3.8	High

System/Evolution Name:
Loss of Reactor Coolant Makeup

Category Statement:
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup:

KA Statement:
How long PZR level can be maintained within limits

UserID: Topic

Question Stem:

Unit-1 is at 20% and stable
Tave is on program
PZR level control is in AUTO
Letdown flow is 75 gpm
Total RCP Leakoff flow is 16 gpm
The running CV pump impeller degrades resulting in a Charging Header flow stablizing at 59 gpm

With continued operation and no operator action, when would Pressurizer level BEGIN to RISE?
ASSUME 130 gallons per % in the PZR.

A 30 -35 minutes

B 40 -45 minutes

C 50 - 55 minutes

D 60 - 65 minutes

Answer: D

Task No:

Obj No: S.RY1-03/20

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RY-XL-01, Ch 14, Pressurizer Reference:

D. is correct because at 20% power, PZR level program is 32%. Level will RISE in the PZR when Letdown Auto isolates at 17%.
Explanation:
32%-17% is 15% level drop. PZR has 130 gal/% ~ 1950 gallons. Level is dropping at 16 (75-59) gpm + 16 gpm RCP leakoff which = 32 gpm. At 32 gpm, 1950 gal will take between 60 and 65 minutes to isolate letdown, then charging will cause PZR level to RISE.

Date Written: 2/28/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
5	Both	1	1	000026	AK3.04	3.5	3.7	Low

System/Evolution Name:
Loss of Component Cooling Water (CCW)

Category Statement:
Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water:

KA Statement:
Effect on the CCW flow header of a loss of CCW

UserID:

Topic

Question Stem:

Component Cooling Water (CCW) is in a normal at power lineup with Unit-1 CCW providing cooling for the
inservice Spent Fuel Pool heat exchanger. A tube leak develops in the Spent Fuel Pool heat exchanger.

Unit-1 CCW Surge Tank level will___(1)____ and CCW pump discharge header flowrate will ___(2)____.

	(1)	(2)
A	rise	drop
B	rise	rise
C	drop	drop
D	drop	rise

Answer: D

Task No:

Obj No: S.CC1-18

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-CC-XL-01, Ch 19, Component Cooling Water System, Pgs 12, 30
1BOA PRI-6, Component Cooling Water Malfunction

Explanation:
CCW is at a higher pressure in the SFP heat exchanger, therefore Surge Tank level will drop and the increased system demand will cause pump discharge flowrate to rise.

Date Written: 2/28/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
6	Both	1	1	000029	2.2.25	2.5	3.7	Low

System/Evolution Name:
Anticipated Transient Without Scram (ATWS)

Category Statement:
Equipment Control

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

UserID: Topic

Question Stem:

10 CFR 50.62 established requirements for reducing the risk of an Anticipated Transient Without Scram (ATWS) in Light-Water-Cooled Nuclear Power Plants.

The ATWS Rule, when applied, is based on reducing the risk of which of the following?

A Exceeding DNBR limit

B Excessive RCS overpressure

C Exceeding Kw/ft Fuel limit

D Excessive RCS cooldown

Answer: B

Task No:

Obj No: S.RP3-01/04

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Byron UFSAR Ch 15.8

Reference:
Ch 60c, I1-RP-XL-03, ATWS Mitigation System

Explanation:

B. is correct because AMS provides a diverse start signal for AFW and a Turbine Trip to maintain the SGs as a Heat Sink to minimize the RCS overpressure transient that occurs with no Reactor Trip.
A. is incorrect since pressure rise precludes reaching DNB.
C. is incorrect since temperature rise adds negative reactivity (Doppler) and limits linear heat rate.
D. is incorrect since a heatup is the concern, cooldown would limit the overpressurization.

Date Written: 2/28/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
7	Both	1	1	000038	EA2.01	4.1	4.7	High

System/Evolution Name:
Steam Generator Tube Rupture (SGTR)

Category Statement:
Ability to determine and interpret the following as they apply to a SGTR:

KA Statement:
When to isolate one or more S/Gs

UserID: Topic

Question Stem:

An accident is in progress on Unit 1.
The following plant conditions exist:

Containment Pressure = 3 psig (slowly dropping)
SG levels (NR): (All are slowly rising)
1A: 5%
1B: 7%
1C: 8%
1D: 5%
Main Steamline 1B radiation ALERT alarm is lit.

In accordance with 1BEP-3, Steam Generator Tube Rupture, the operator is directed to:

A immediately manually CLOSE the 1B AF isolation valves, 1AF013B and F.

B maintain feed to the 1B SG until narrow range level is 10%, then manually isolate AF to the 1B SG.

C maintain feed to all SGs until all narrow range levels are at or above 10%, then manually isolate AF to the 1B SG.

D maintain feed to 1B SG until narrow range level is 31%, then manually isolate AF to the 1B SG.

Answer: B

Task No:

Obj No: T.EP04-08

Question Source: Byron 2000 NRC exam

Question Difficulty: Medium

Time: 1

Cross Ref:
1/2BEP-3, Steam Generator Tube Rupture, Step 4
I1-EP-XL-04, BEP-3 series, Pg 15

Reference:

Explanation:
The ruptured SG level must be > 10% to ensure an adequate thermal layer exists prior to isolation of Aux Feed. This will insulate the steam space in the SG from the cooler RCS water as the RCS is cooled down to allow depressurization of the RCS to stop the leakage to the SG without losing subcooling, if the SG were to depressurize while trying to equalize RCS to SG pressure.
A. is incorrect since the >10% is required prior to isolation.
B. is correct as described above.
C. is incorrect since the requirement for isolation only applies to the ruptured SG. The other, intact, SGs can be throttled for heat removal control once at least one of them is >10%.
D. is incorrect since this is the ADVERSE CNMT value which is not used when CNMT pressure is < 5 psig for these conditions.

Date Written: 2/27/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
8	Both	1	1	000040	AK2.02	2.6	2.6	High

System/Evolution Name:
Steam Line Rupture

Category Statement:
Knowledge of the interrelations between the Steam Line Rupture and the following:

KA Statement:
Sensors and detectors

UserID: Topic

Question Stem:

Unit 1 tripped from 100% power due to a steamline break inside of Containment. Shortly after the trip, the following parameters were recorded:

PZR pressure is 1750 psig and stable
PZR level is 22% and stable
CNMT pressure has reached 7.8 psig
S/G NR levels; 1A 31%, 1B 30%, 1C 25%, 1D 34%
S/G pressures; 1A 760 psig, 1B 775 psig, 1C 660 psig, 1D 800 psig

An automatic steamline isolation occurred due to_________.

A low S/G pressure

B low S/G level

C containment pressure circuit

D low PZR pressure SI

Answer: A

Task No:

Obj No: S.MS1-07-D

Question Source: 2000 Byron NRC exam

Question Difficulty: Low

Time: 1

Cross Ref:

I1-MS-XL-01, Ch 23, Main Steam System

Reference:

Explanation:

A. is correct since a steamline break would cause steamline pressure to drop rapidly. The low pressure setpoint is 640 psig, but is rate sensitive and could have occurred without actually going below 640 psig.
B. is incorrect because the SG level circuit does not input to the steamline isolation logic.
C. is incorrect since 8.2 psig in CNMT has not been reached.
D. is incorrect since low PZR pressure SI does not cause a main steamline isolation to occur.

Date Written: 3/2/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
9	Both	1	1	000054	AA1.03	3.5	3.7	High

System/Evolution Name:
Loss of Main Feedwater (MFW)

Category Statement:
Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater (MFW):

KA Statement:
AFW auxiliaries, including oil cooling water supply

UserID: Topic

Question Stem:

Unit 1 was at 100% power.
Unit 2 is in MODE 6.
2B SX pump is OOS for replacement

A spurious Feed Water Isolation occurred on Unit 1. The crew manually tripped the Unit 1 Reactor. When the Unit 1 Main Generator output breakers opened, Off-Site power was lost to BOTH Units resulting in the following condition:

1A DG started and loaded. 1A SX pump tripped after starting.
1B DG did not start
2A DG is running, but the output breaker will NOT CLOSE.
2B DG started and loaded

What is the IMPACT on CONTINUED operation of the Unit 1 Aux Feed Pumps?
(Ignore any fuel consumption concerns)

A Both pumps can continue to run as long as Fuel Oil is available.

B 1A only requires 1A DG running; 1B will trip on High Bearing temperature.

C 1B will trip on High Jacket Water temperature; 1A can continue to run.

D 1B can continue to run as long as Fuel Oil is available; 1A should be stopped to prevent tripping.

Answer:	Task No:		Question Source:	Question Difficulty
D	Obj No:	S.AF1-04/15	New	Medium
Time: 1	Cross Ref:			

I1-AF-XL-01, Ch 26, Auxiliary Feedwater System Reference:

Explanation:

A. is incorrect because 1A requires an SX pump running for oil cooling and will rapidly overheat bearings and most likely trip the supply breaker.
B. is incorrect because of the reasoning in A. above. It may run until the EDG overheats and stops, but will probably have already tripped. 1B is designed to run without an SX pump running. It has it's own shaft driven SX pump.
C. is incorrect for the same reasons described in B. above.
D. is correct because 1B has it's own shaft driven SX pump and will supply cooling as long as an SX reservoir is maintained while 1A requires an SX pump running to provide oil cooling and should be stopped when the no SX condition is recognized to prevent damage beyond repair.

Date Written: 3/2/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
10	Both	1	1	000055	EK1.01	3.3	3.7	Low

System/Evolution Name:
Loss of Offsite and Onsite Power (Station Blackout)

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the Station Blackout:

KA Statement:
Effect of battery discharge rates on capacity

UserID: Topic

Question Stem:

Unit 1 is at 100% power
- At 0800 a Loss of all AC power occurs on Unit 1.
- DC Bus 111 and 112 is being supplied by their respective Batteries.

With no operator action, the MINIMUM DESIGN DC Bus voltage will only be available on Unit 1 DC Buses until_______?

A 0830

B 0900

C 1000

D 1600

Answer: B

Task No:

Obj No: S.DC1-04-B/05-C

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-DC-XL-01, Ch 8a, 125 VDC Power Systems
Byron UFSAR, CH 8.3
1BOA ELEC-3, Loss of 4KV ESF Bus

Reference:

Explanation:

The design of the 1E 125 VDC Batteries is Blackout with LOCA and only one DG (no AC to Battery Charger for 1 hour) and minimum voltage of 105 VDC. 1BOA ELEC-3 states that if a battery charger cannot be restored and the opposite unit charger is energized, then crosstie within 1 hour per BOP DC-7.
A. is incorrect but a plausible action time.
B. is correct as described above.
C. is incorrect but plausible action time used in Tech Specs for electrical problems.
D. is incorrect but plausible action time used in Tech Specs for electrical problems.

Date Written: 3/2/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
11	Both	1	1	000056	AK1.04	3.1	3.2	High

System/Evolution Name:
Loss of Offsite Power

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power:

KA Statement:
Definition of saturation conditions, implication for the systems

UserID: Topic

Question Stem:

A Reactor Trip with a Loss of Off-Site Power has occurred. The following conditions exist:

CETCs = 582°F and stable
All Hot Leg Temperatures = 580°F and stable
All Cold leg Tempertures = 550°F and stable
Pressurizer Pressure = 2085 psig and stable

It is desired to reduce RCS pressure but maintain 50°F subcooling. The MINIMUM pressure that will maintain 50°F RCS subcooling as indicated on SPDS is ____.

A 1916 psig

B 1928 psig

C 1933 psig

D 1948 psig

Answer: C

Task No:

Obj No: T.EP01-06-D

Question Source: Byron 2001 Cert exam

Question Difficulty: Medium

Time: 1

Cross Ref:

1BEP 0.2, Natural Circulation Cooldown, Fig 1BEP 0.2-2 and ATT A
I1-EP-XL-01, 1BEP-0, Reactor Trip or Safety Injection - series pgs 95, 100
Steam Tables

Reference:

Add 50°F to 582°F (highest temp) = 632°F; look up Sat Press for 632°F in Steam Tables = 1947 psia; subtract 15 psi = 1932 psig

Explanation:

A. Is incorrect but plausible if wrong temp used and subtracted 15psi instead of adding for steam table use.
B. is incorrect but plausible if 15 psi was not added in for steam table use.
C. is correct as described above.
D. is incorrect but plausible if 15 psi was not subtracted back out of the steam table result to get psig.

Date Written: 3/2/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
12	Both	1	1	000057	AK3.01	4.1	4.4	High

System/Evolution Name:
Loss of Vital AC Electrical Instrument Bus

Category Statement:
Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus:

KA Statement:
Actions contained in EOP for loss of vital ac electrical instrument bus

UserID: Topic

Question Stem:

Unit 1 was at 100% power when a Reactor Trip and Safety Injection occurred due to a SGTR.
2 Minutes later, Off-site power was lost.
Both 1A and 1B DG started and closed on to their respective buses.
All B Train loads are running.
A Train loads are being manually started.

When the 1A AF pump is started, the operator notices that the flowrates to each SG from the 1A AF pump rapidly LOWER from ~ 180 gpm to 0 gpm.

Of the following, the manual load start and the resultant 1A AF flow response occurred because of loss of ____________.

A 125 VDC Bus 111

B 120 VAC Bus 111

C 125 VDC Bus 113

D 120 VAC Bus 113

Answer: B

Task No:

Obj No: S.AP1-14-B

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:
1BOA ELEC-2, Loss Of Instrument Bus Unit 1
I1-AP-XL-01, AC Electrical Power Systems

Reference:

Explanation:
Sequencer for A Train equipment and to control setpoint signal for the A Train 1AF005 valves are powered by 120 VAC Bus 111 requiring manual sequencing of loads and causes the flow control setpoint for the 1AF005 valves to position for a zero flow setpoint.
A. is incorrect because this would not allow breaker operation of A Train loads from the MCR.
B. is correct as stated above.
C. is incorrect because this would have no impact on sequencing of loads or AF valve control.
D. is incorrect because this would also have no impact on sequencing of loads or AF valve control.

Date Written: 3/3/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
13	Both	1	1	000058	AA1.01	3.4	3.5	High

System/Evolution Name:
Loss of DC Power

Category Statement:
Ability to operate and/or monitor the following as they apply to the Loss of DC Power:

KA Statement:
Cross-tie of the affected dc bus with the alternate supply

UserID: Topic

Question Stem:

At 0800:

- 125 VDC Bus 111 is being supplied by Battery 111 due to failure of the Battery Charger.
- 125 VDC Bus 111 voltage is 122 VDC.
- 125 VDC Bus 211 is being supplied by it's Battery Charger and voltage is steady at 128 VDC.

At 0805:

- 125 VDC Bus 111 voltage 120 VDC

If the rate of usage has been STEADY from Time = 0, what is the LATEST time BEFORE it becomes UNACCEPTABLE to X-tie with a load on 125 VDC Bus 111?

A 0816

B 0822

C 0830

D 0835

Answer: D

Task No:

Obj No: S.DC1-05-D

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

BOP DC-7, 125 VDC ESF Crosstie/Restoration
I1-DC-XL-01, Ch 8a, 125 VDC Power Systems

Reference:

Explanation:

BOP DC-7 states in a Precaution that X-tie to a loaded DC Bus should not occur with > 20 volts differential. At 2 VDC usage every 5 minutes, it will take 20 minutes additional time to reach 108 VDC on Bus 111, so at 21 minutes the 20 VDC differential limit will be exceeded.

A. is incorrect since Bus 111 voltage will be between 112 and 114 and it is still aceptable to crosstie.
B. is incorrect since Bus 111 voltage will be between 111 and 112 and it is still aceptable to crosstie.
C. is incorrect since Bus 111 voltage will be 110 and it is still aceptable to crosstie
D. is correct since Bus 111 voltage will be at 108 and any more time will be > 20 volts difference.

Date Written: 3/4/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
14	Both	1	1	000062	AK3.04	3.5	3.7	High

System/Evolution Name:
Loss of Nuclear Service Water

Category Statement:
Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:

KA Statement:
Effect on the nuclear service water discharge flow header of a loss of CCW

UserID: Topic

Question Stem:

Unit 1 and Unit 2 are at 100% power in a normal at power lineup with the following conditions:

- 2B SX pump is running.
- 2A CC pump has tripped on overcurrent.
- 2B CC pump has auto started.
- A large tube leak develops in the U-2 CC Heat Exchanger.
- U-2 CC Surge Tank level is dropping rapidly with full make-up capacity.

Considering the impact on 2B SX pump alone, with no operator action, what would be the response of 2B
SX pump discharge pressure, AFTER the U-2 CC Surge Tank level drops to 13%, and why?

A Discharge pressure will initially drop due to the 2B CC pump trip, then rise and stablize below the original value.

B Discharge pressure will rise immediately due to the loss of the large heat load and remain above the original value.

C Discharge pressure will drop and stablize at a lower value due to the loss of the large heat load.

D Discharge pressure will rise due to the 2B CC pump trip, then drop due to the loss of the large heat load and should stablize close to the original value.

Answer: A

Task No:

Obj No: S.SX1-15, S.CC1-18

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-SX-XL-01, Ch 20, Essential Service Water System,
I1-CC-XL-01, Ch 19, Component Cooling Water System, Pg 4, 12, 26

Explanation:
CC is at a higher pressure than SX. The SX pump discharge pressure is normally ~ 90 psig. The CCW pump discharge pressure is ~ 130 psig with system reliefs are set at 150 psig or higher. When the 2B CC pump trips, the SX pump discharge will drop initially while the CC system is refilled by SX. When CC is refilled, a very small demand will continue through the Surge tank vent. CCW heat load on SX is minimal with both Units at 100% power.
A. is correct because pressure drops originally due to the increase demand on the system to refill CCW. Once CCW is full, the increased demand will be small through the surge tank vent, thus the pressure will recover, but remain below the original value.
B is incorrect because SX flow goes up, thus pressure drops.
C. is incorrect because the completion of CCW refill will cause pressure to rise.
D. is incorrect because the increased SX flow will cause pressure to drop.

Date Written: 3/12/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
15	Both	1	1	000065	AA1.02	2.6	2.8	High

System/Evolution Name:
Loss of Instrument Air

Category Statement:
Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air:

KA Statement:
Components served by instrument air to minimize drain on system

UserID: Topic

Question Stem:

Unit 2 has completed refueling and is in the process of plant heatup with the following conditions:

- 2B RH is in shutdown cooling mode.
- RCS temperature is 300°F.
- RCS pressure is 340 psig.
- 2B CV pump is in operation.
- PZR bubble has just been established.
- A loss of instrument air has just occurred.

Which ONE of the following describes the INITIAL response and WHY, if NO operator action is taken?

A RCS will cooldown due to the RISE in RH flow through the 2B RH Hx.

B PZR level DROPS due to the RISING RH letdown flow.

C 2B CV pump suction pressure will DROP due to the RISE in charging flow.

D RCS will heatup due to the DROP in RH flow through the 2B RH Hx.

Answer:	Task No:		Question Source:	Question Difficulty
A	Obj No:	S.RH1-11, T.OA39-03	Byron NRC exam bank (1996)	Medium

Time: 1 Cross Ref:

Reference:
Ch 18, Residual Heat Removal, Pg 12
1BOA SEC-4, Loss of Instrument Air, Table A, Pg 12
I1-OA-XL-39, Loss of Instrument Air, Pg 31

Explanation:
B Train RHR Hx outlet valve, 1RH607, fails open and the Hx bypass, 1RH619, fails closed forcing total flow through the RHR Hx.
CCW flow through the Hx remains unchanged, therefore an RCS cooldown will occur.
A. is correct because of the rise in RH flow through the Hx.
B. is incorrect because RH letdown flow stops due control valve 1CV-131 failing closed and PZR level will go up with charging to the RCP seals maintained.
C. is incorrect because total charging flow will drop and CV pump suction pressure will rise or stay about the same.
D. is incorrect because a cooldown will occur due to valve fail positions in th RH system.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
16	Both	1	1	00WE04	EK2.2	3.8	4.0	Low

System/Evolution Name:
LOCA Outside Containment

Category Statement:
Knowledge of the interrelations between the LOCA Outside Containment and the following:

KA Statement:
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

UserID: Topic

Question Stem:

A small break LOCA has occurred on Unit 1 outside containment.

Actions of 1BCA-1.2, LOCA Outside Containment, have been completed.
- RCS pressure continues to DROP.

A transition was made to 1BCA-1.1, Loss of Emergency Coolant Recirculation.

Which of the following is the reason a transition was made to 1BCA-1.1?

A To recover after the break was isolated.

B To terminate off-site dose release.

C To reverify that all automatic actions have been completed.

D To take compensatory actions for lack of inventory in the contaiment sump.

Answer: D

Task No:

Obj No: T.CA2-05

Question Source: Byron NRC bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
1BCA-1.1, Loss of Emergency Coolant Recircualtion
1BCA-1.2, LOCA Outside of Containment
I1-CA-XL-02, BCA 1.1, 1.2 Contingency Action, Pg 1

Explanation:
This procedure is used when recirculation cannot be accomplished due to alignment problems or lack of inventory in the CNMT sump and tries to delay depletion of the RWST by stopping unnecessary ECCS and CS pumps and reducing RCS pressure to slow or stop the leakage.
A. is incorrect since RWST inventory and leakage would no longer be a concern.
B. is incorrect since this BCA is not intended to isolate a barrier.
C. is incorrect since these actions would have been addressed in other procedures and this BCA is strictly buying time for inventory depletion.
D. is correct as described above.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
17	Both	1	1	00WE11	EK2.1	3.6	3.9	High

System/Evolution Name:
Loss of Emergency Coolant Recirculation

Category Statement:
Knowledge of the interrelations between the Loss of Emergency Coolant Recirculation and the following:

KA Statement:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

UserID: Topic

Question Stem:

Given the following conditions on Unit 1:

- Reactor Trip and SI occurred at 0700.
- RH system problems resulted in a loss of recirculation capability.
- Current time is 1340
- RCS subcooling is 10°F
- 1A and 1B CV pumps are running
- Hi head SI flow is 350 gpm (Assume equal flow from each CV pump)
- 1A SI pump discharge flow is 20 gpm
- 1B SI pump discharge flow is 50 gpm

In order to meet the MINIMUM ECCS flow for decay heat removal, the ONLY ECCS pump(s) that should be RUNNING will be ________. (Figure 1BCA 1.1-1 is attached)

A BOTH SI pumps

B BOTH CV pumps

C 1A CV pump and 1A SI pump

D 1A CV pump and 1B SI pump

Answer: D

Task No:

Obj No: T.CA2-05

Question Source: Byron Cert bank (2001)

Question Difficulty: Low

Time: 1

Cross Ref:

Reference:
1BCA 1.1, Loss of Emergency Coolant Recirculation, Step 15 RNO
I1-CA-XL-02, BCA 1.1, 1.2 Contingency Action, Pg 17

Explanation:
Per Figure BCA 1.1-1, the required flow for 400 minutes (6 hours, 40minutes) is ~ 220 gpm. To accomplish this and delay RWST level drop, 1 CV pump with 175 gpm/pump and the 1B SI pump with 50 gpm would be the closest combination given to provide adequate decay heat removal and minimize the rate of RWST level drop.
A. is incorrect since required flow rate is at least 220 gpm and this would only provide 70 gpm.
B. is incorrect since 350 gpm will provide adequate decay heat removal, but excessive depletion of the RWST.
C. is incorrect since the flow rate of 195 gpm is not adequate for the ~ 220 gpm required for decay heat removal.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref: Figure 1BCA-1.1-1

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
18	Both	1	1	00WE05	2.1.27	2.8	2.9	Low

System/Evolution Name:
Loss of Secondary Heat Sink

Category Statement:
Conduct of Operations

KA Statement:
Knowledge of system purpose and or function.

UserID: Topic

Question Stem:

The first step of 1BFR-H.1, Response to Loss of Secondary Heat Sink, is to check if RCS pressure is greater than any intact SG pressure. If NOT, the crew is sent to 1BEP-1 or 1BEP ES-1.3.

Why is 1BFR-H.1 NOT performed under these conditions?

A ECCS injection is NOT providing adequate cooling.

B The SGs are NOT needed as a sink.

C Higher priority RED paths are anticipated due to the indicated large break LOCA.

D Cold leg recirculation, re-establishes thermal coupling with the SGs.

Answer: B

Task No:

Obj No: T.FR03-03

Question Source: Ginna NRC bank (1996)

Question Difficulty: Medium

Time: 1

Cross Ref:

1BFR-H.1, Response to Loss of Secondary Heat Sink. Reference:
I1-FR-XL-03, BFRs H.1-H.5, Pg 5

Explanation:

With the RCS depressurized below the SGs, the break size is large enough for ECCS to provide adequate heat removal. Also, when this occurs, the SGs are no longer thermodynamically coupled and they become a heat source instead of a heat sink, therefore, actions to restore them as a heat sink are not required and not necessary.
A. is not correct for all LOCAs
B. is correct for this assessment.
C. is not correct, assumed ECCS is adequate. Higher order RED paths should NOT be anticipated with adequate ECCS flow.
D. is incorrect since ECCS flow should be adequate before cold leg recirc is required.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
19	Both	1	2	000024	AK3.02	4.2	4.4	High

System/Evolution Name:
Emergency Boration

Category Statement:
Knowledge of the reasons for the following responses as they apply to the Emergency Boration:

KA Statement:
Actions contained in EOP for emergency boration

UserID: Topic

Question Stem:

Given the following Unit 1 plant conditions:

- Reactor Tripped 20 minutes ago due to a partially stuck open SG safety valve.
- 2 RCCAs from Shutdown Bank B stuck in the mid-out position.

- A loss of off-site power occurred while completing Step 3 of 1BEP-0, REACTOR TRIP OR SAFETY INJECTION
- 1B DG is OOS
- 1A DG is operating as expected.

- RCS temperature is currently 548°F.
- The stuck open SG safety is causing a cooldown rate of 15°F/Hour.

Which of the following is/are required operator action(s) and why?

A Emergency borate using 1B CV pump from the RWST and MAXIMIZE CV pump flow due to the 2 RCCAs NOT fully inserted.

B Emergency borate using the Boric Acid Transfer pump due to the cooldown and the 2 RCCAs NOT fully inserted.

C No action is required because the Loss of Offsite Power ensures ALL Rods are fully inserted.

D Emergency borate using the 1A CV pump from the RWST and MAXIMIZE charging flow due to the cooldown and 2 RCCAs NOT fully inserted.

Answer:	Task No:		Question Source:	Question Difficulty
D	Obj No:	T.EP01-06-C,	Byron NRC exam bank (2000)	Medium
Time:	Cross Ref:			
1				

Reference:
1BEP ES-0.1, Reactor Trip or Safety Injection
I1-EP-XL-01, BEP-0, Reactor Trip or Safety Injection, Pgs 57, 60
BOA PRI-2, Emergency Boration
I1-OA-XL-13, BOA PRI-2, Emergency Boration, Pg 1

Explanation:
BEP ES-0.1 requires emergency boration for RCS temperature < 557°F and for each rod not fully inserted if more than 1 is not fully inserted. Emergency boration can be accomplished from the BATs via Boric Acid Transfer pump to the CV pump suction or from the RWST via the CV pump. For this case, only 1A CV pump is available. Boric Acid Transfer pump is non ESF power.
A. is incorrect since 1B CV pump has no power.
B. is incorrect since the flowpath must include a CV pump (1A).
C. is incorrect since Stuck RCCAs are still Stuck and the cooldown is still in progress.
D. is correct as described.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
20	Both	1	2	000032	2.2.22	3.4	4.1	High

System/Evolution Name:
Loss of Source Range Nuclear Instrumentation

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID: Topic

Question Stem:

The following conditions exist on Unit 1:

- A reactor startup is in progress.
- Source range counts are 6.5E4 cps on N-31 and 6.95E4 on N-32.
- Intermediate range power is 3.2E-5% on N-35 and 4.1E-5% on N-36.
- Control Bank D is at 136 steps.

Just prior to blocking SR NI's, N-31 fails low.

Which of the following identifies the power limits due to the SR N-31 failure?

A Power must be immediately reduced to < P-6.

B NO further positive reactivity additions can occur.

C Reactor startup may proceed.

D Power may be raised, but a MODE change SHALL NOT occur.

Answer: C

Task No:

Obj No: T.OA10-06/12, S.NI1-

Question Source: Byron 2001 Cert exam

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
BOA INST-1, Nuclear Instrument Malfunction
I1-OA-XL-10, BOA INST-1, NI Malfunction, Pg 13
Tech Spec/Bases 3.3.1, RTS Instrumentation
I1-NI-XL-01, Ch 31, Source Range Nuclear Instrumentation, Pg 23

Explanation:
A trip will not occur with the SR channel failing low. The Tech Spec applies in MODE 2 with power < P-6. The condition given is > P-6 and adequate IR overlap is given, therefore Blocking SR would be appropriate and power ascension may continue.
A. is incorrect since this is not required action.
B. is incorrect since power is > P-6, if below P-6, this would be required.
C. is correct as described above.
D. is incorrect, this is not a MODE change requirement.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
21	Both	1	2	000059	AK2.01	2.7	2.8	Low

System/Evolution Name:
Accidental Liquid Radwaste Release

Category Statement:
Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:

KA Statement:
Radioactive-liquid monitors

UserID:

Topic

Question Stem:

What action occurs when 0RE-PR16J, 0A Blowdown After Filter Outlet Radiation Monitor, detects a high radiation condition?

A The inlet valve to the Blowdown Monitor tank CLOSES, then re-opens automatically when the high radiation condition clears.

B The isolation valve to the main condenser or CST CLOSES, then re-opens automatically when the high radiation condition clears.

C The inlet valve to the Blowdown Monitor tank OPENS, then must be manually closed when the high radiation condition clears.

D The isolation valve to the main condenser or CST OPENS, then must be manually closed when the high radiation condition clears.

Answer: C

Task No:

Obj No: S.AR1-04-B-04

Question Source: Byron NRC exam bank (2000) - Modified

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-AR-XL-01, Ch 49, Radiation Monitors, Pgs 28
BAR RM-11 for 0RE-PR16J

Reference:

Explanation:

When 0RE-PR16J generates a HIGH alarm the auto action will close isolation valve to the condenser or CST after opening the inlet to the Blowdown Monitor Tank. This allows collection of radioactive drains in a tank that can be aligned for processing in radwaste. The interlock is only auto in this direction and requires manual realignment after the condition clears. This will prevent a monitor malfunction from causing a realignment resulting in a release to the environment.
A. is incorrect since this valve is opened and no auto realignment takes place.
B. is incorrect since no auto realigment occurs.
C. is correct as described above.
D. is incorrect since this valve is closed, not opened.

Date Written: 3/5/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
22	Both	1	2	000061	AK1.01	2.5	2.9	Low

System/Evolution Name:
Area Radiation Monitoring (ARM) System Alarms

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms:

KA Statement:
Detector limitations

UserID: Topic

Question Stem:

Which of the following transients will cause Main Steam Line Radiation Monitors, _AR22A/B/C/D, to indicate ERRONEOUSLY?

A LOCA inside Containment

B Steam Generator Tube Rupture

C Main Steam Line break inside Containment.

D Feed Line break in the Safety Valve Enclosure.

Answer: D

Task No:

Obj No: A.PF3-03, S,AR1-02-

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-PF-XL-03, I&C Ch 3, Radiation Detection and Measurement
I1-AR-XL-01, Ch 49, Radiation Monitors, Pgs 7
BAR RM11-1-1AR22J

Explanation:
These monitors use GM detectors and are placed in close proximity to each of the Main Steam lines in the Safety Valve Enclosure. Since they are proximity devices, they have very little shielding around them from the room environment and, as stated in the BAR, are succeptible to alarming due to high temperature in the area.
A. is incorrect since the harsh environment is in CNMT and not in the vicinity of the detectors.
B. is incorrect since the detectors should work as designed for this condition, i.e. no change to their environment.
C. is incorrect since the harsh environment is in CNMT and not in the vicinity of the detectors.
D. is correct since this is the same enclosure where the detectors are located and high temperatures would result.

Date Written: 2/27/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
23	Both	1	2	000068	AA2.11	4.3	4.4	High

System/Evolution Name:
Control Room Evacuation

Category Statement:
Ability to determine and interpret the following as they apply to the Control Room Evacuation:

KA Statement:
Indications of natural circulation

UserID: Topic

Question Stem:

Given the following conditions on Unit 1:

- Reactor tripped due to a fire in the Upper Cable Spreading Room.
- All RCPs are stopped
- The crew is in 1BOA PRI-5, Control Room Inaccessability Unit 1, with a cooldown established from the Remote Shutdown Panel.

Which of the following indicates core heat removal by natural circulation is DEGRADING? (Consider each condition independently)

A SG pressures have dropped from 700 psig to 600 psig and are now starting to drop at a slower rate.

B PZR level is 15% and slowly dropping while indicated subcooling is 80°F and rising.

C Thot is 520°F and Tcold is 460°F and the Delta T between them is rising.

D Thot is 490°F and dropping slowly and indicated CETCs are dropping slowly.

Answer: C

Task No:

Obj No: A.HT5-03/04

Question Source: Byron bank

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-HT-XL-01, Ch 5, Natural Circulation, Pg 11
BOA PRI-5, Control Room Inaccessability, Step 18 RNO

Explanation:
Items verified for adequate Nat Circ flow are subcooling, CETCs stable or dropping, Thot stable or dropping and consistent with CETCs (CETCs will normally be slightly higher), Tcold ~ Tsat for SG pressure, and SG pressure stable or dropping. One other consideration is the design Nat Circ power-to-flow ratio is such that Thot-Tcold (delta-T) should never be > full power delta-T (60°F).
A. is incorrect since this meets the Nat Circ verification criteria.
B. is incorrect since subcooling is adequate and PZR level response is not a criteria.
C. is correct since the delta-T is 60°F and rising. This denotes the driving head of Tcold is not adequate to provide sufficient flow through the core.
D. is incorrect since this is an adequate response to verify Nat Circ is occurring.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
24	Both	1	2	00WE14	EA2.2	3.3	3.8	Low

System/Evolution Name:
High Containment Pressure

Category Statement:
Ability to determine and interpret the following as they apply to the High Containment Pressure:

KA Statement:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

UserID: Topic

Question Stem:

Given the following conditions on Unit 1:

- A LOCA has occurred.
- Transition has been made to 1BEP ES-1.3, Transfer to Cold Leg Recirculation.
- Containment Spray (CS) actuated as designed.
- Spray Add Tank LO-2 lights are LIT
- All systems are operating as expected.

When and how is Containment Spray terminated?

A ONE CS pump is stopped when containment pressure is < 15 psig. The other CS pump is stopped when RWST LO-3 level is reached.

B ONE CS pump is stopped when containment pressure is < 20 psig. The other CS pump is stopped after it has operated for at least 2 hours.

C BOTH CS pumps are stopped when containment pressure is < 15 psig and have operated for at least 2 hours.

D BOTH CS pumps are stopped when containment pressure is < 20 psig and RWST LO-3 level is reached.

Answer:	Task No:		Question Source:	Question Difficulty
C	Obj No:	S.CS1-12	Byron NRC exam bank (1998)	Medium

Time: 1

Cross Ref:

Reference:
Ch 59, Containment Spray System, Pg 29
1BEP-1, Loss of Reactor or Secondary Coolant, step 7

Explanation:
Following a LOCA it is part of the design criteria to terminate CS when containment pressure has dropped below 15 psig and the Spray Add tank contents have been injected into the CNMT atmosphere and sump and recirculatedfor 2 hours for iodine removal and maintenance.
A. is incorrect since no CS pumps are required to be run once CNMT pressure falls below 15 psig, however, 2 hours is required. If necessary, at Lo-3 in the RWST, suction is switched to the CNMT sump.
B. is incorrect since the criteria is < 15 psig and both pumps are stopped.
C. is corrrect as described above.
D. is incorrect since the criteria is < 15 psig and 2 hours. If necessary, at Lo-3, suction is switched to the CNMT sump.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
25	Both	1	2	00WE06	EA1.3	3.7	4.0	Low

System/Evolution Name:
Degraded Core Cooling

Category Statement:
Ability to operate and/or monitor the following as they apply to the Degraded Core Cooling:

KA Statement:
Desired operating results during abnormal and emergency situations

UserID: Topic

Question Stem:

A CAUTION in 1BFR-C.2, Response to Degraded Core Cooling, states that an SI accumulator injection may cause a Red path condition in INTEGRITY and that 1BFR-C.2 should be completed before transition to 1BFR-P.1, Response to Imminent Pressurized Thermal Shock Condition. The CAUTION applies during depressurization, prior to transitioning to 1BFR-P.1.

What is the reason for this CAUTION?

A Responding to the INTEGRITY Red path at this time could result in a CORE COOLING Red path.

B The INTEGRITY Red path only protects the RCS barrier and the continued actions in 1BFR-C.2 will protect both the fuel clad and the RCS barriers.

C Responding to the INTEGRITY Red path at this time could result in an INVENTORY Red path.

D The INTEGRITY Red path will be corrected by continuing the actions of 1BFR-C.2.

Answer: A

Task No:

Obj No: T.FR02-02

Question Source: Byron NRC bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref: I1-FR-XL-02, BFR-C.1, C.2, C.3, Pg 54

Reference: BFR C.2, Response to Degraded Core Cooling, step 10

Explanation:

SI Accumulator injection will cause a significant temperature drop and rapid cooldown and may cause a Red path on INTEGRITY, however, exiting before completing BFR C.2 would result in only a short term cooldown resulting in a Red path on CORE COOLING, since the first thing BFR-P.1 would have you do is stop the cooldown.
A. is correct as described above.
B. is incorrect since INTEGRITY is directly concerned with the RCS barrier and CORE COOLING action here is trying to cool the fuel to minimize/prevent fuel barrier damage and is not concerned at this point in protecting the RCS barrier.
C. is incorrect since CORE COOLING is a higher priority and INVENTORY at worst can be a Yellow path.
D. is incorrect since this is not necessarily true, but for barrier protection of the fuel, it is more important to gain control of core cooling, then deal with INTEGRITY.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
26	Both	1	2	00WE03	EK3.1	3.3	3.7	Low

System/Evolution Name:
LOCA Cooldown and Depressurization

Category Statement:
Knowledge of the reasons for the following responses as they apply to the LOCA Cooldown and Depressurization:

KA Statement:
Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics

UserID: Topic

Question Stem:

A reactor trip and SI have occurred on Unit 2.

- Control room operators are responding to a small break LOCA.
- All RCPs are stopped.
- Containment pressure is normal.
- The crew has transitioned to 2BEP ES-1.2, Post-LOCA Cooldown and Depressurization.
- A PZR PORV is being used to depressurize the RCS until PZR level is > 25%.

In addition to ensuring that RCS conditions are under adequate operator control, what is the basis for establishing this PZR level?

A Ensures sufficient inventory to prevent a low PZR level condition when a RCP is started.

B Ensures that a reduction in subcooling does not occur when SI flow is reduced.

C Ensures that letdown can be established prior to starting a RCP.

D Ensures adequate PZR steam space to absorb pressure fluctuations during an RCP start.

Answer: A

Task No:

Obj No: T.EP02-01-D

Question Source: Byron NRC bank (1996)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-EP-XL-02, Loss of Reactor or Secondary Coolant, Pgs 86, 87. Reference:
1/2BEP ES-1.2, Post-LOCA Cooldown and Depressurization

Explanation:

This level is established to start one RCP. It is assumed the level will drop when the RCP is started and allows adequate level to maintain PZR level and pressure control to maintain the RCP running as the possible head void is collapsed.
A. is correct as described above.
B. is incorrect since this is not part of the ECCS flow reduction segment.
C. is incorrect since letdown is not a concern at this point. Forced flow for cooldown and pressure control is.
D. is incorrect since the concern is adequate inventory to support the RCP start to ensure it continues to run. There is plenty of steam space and the subsequent start will cause pressure to drop, not rise.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
27	Both	1	2	00WE10	EK2.1	3.3	3.5	Low

System/Evolution Name:
Natural Circulation with Steam Void in Vessel with/without RVLIS

Category Statement:
Knowledge of the interrelations between the Natural Circulation with Steam Void in Vessel with/without RVLIS and the following:

KA Statement:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

UserID: Topic

Question Stem:

Given the following Unit 1 conditions:

- The crew is in 1BEP ES-0.2, Natural Circulation Cooldown.
- PZR pressure is being controlled using Aux. Spray and PZR heaters.
- Charging and letdown are in manual and are balanced.

As pressure is being lowered through 1300 psig, a rapid rise is observed in PZR level.

What action is required to be taken?

A Repressurize the RCS.

B Isolate the SI accumulators.

C Raise the RCS cooldown rate.

D Place excess letdown in service.

Answer: A

Task No:

Obj No: T.EP01-06-D

Question Source: Byron NRC bank (1998)

Question Difficulty: Medium

Time: 1

Cross Ref:

1BEP ES-0.2, Natural Circulation Cooldown step 15 RNO
I1-EP-XL-01, Reactor Trip or SI, Pg 102

Reference:

Explanation:

With vessel head cooling available, this procedure assumes no head voiding will occur. At step 15, any substantial rise in PZR level or RVLIS indicating head voiding is occurring requires immediate repressurization of the RCS to collapse the void.
A. is correct as described above.
B. is incorrect since this is done at RCS pressure < 1000 psig as in any normal cooldown and at the pressure in question, SI accumulators would not be the reason for the rapid PZR level rise.
C. is incorrect since the occurrence of head voiding is not diminished by raising the cooldown rate of the RCS. The metal mass in the head has very minimal cooling from the RCS fluid in Nat Circ.
D. is incorrect since additional letdown will remove inventory, but the inventory has not risen, only displaced by an additional steam bubble.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
28	Both	2	1	003000	2.1.33	3.4	4.0	High

System/Evolution Name:
Reactor Coolant Pump System (RCPS)

Category Statement:
Conduct of Operations

KA Statement:
Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

UserID: Topic

Question Stem:

Given the following Unit 1 conditions:

- RCS temperature (Average CETCs) = 340°F
- All SG pressures = 100 psig
- RCS pressure = 390 psig
- RCPs 1B and 1D running
- RCPs 1A and 1C have breakers tagged OOS
- RHR loops 1A and 1B are aligned for ECCS

RCP 1D has just tripped on breaker overcurrent.

What is/are the required action(s)?

A Immediately take actions to place the unit in MODE 5 with either RHR Train in operation.

B Immediately place one train of RHR in service for shutdown cooling.

C Restore 1A or 1C RCP to OPERABLE within 1 hour.

D Immediately restore 1A or 1C RCP to service and make 1A or 1C RCP OPERABLE.

Answer: D

Task No:

Obj No: S.RC1-12

Question Source: Byron NRC exam bank (1998)

Question Difficulty: Medium

Time: 1

Cross Ref:

Byron ITS Section 3.4.6

Reference:
I1-RC-XL-01, Ch 12, Reactor Coolant System

Explanation:

The Tech Spec requires 2 loops of 4(2 RCS + 2 RHR) be OPERABLE and at least ONE in operation in MODE 4. The action requirement (Cond B) states that for one required loop inoperable to initiate action to restore a second loop to operable status immediately.
A. is incorrect since a RHR loop is unavailable for SDC in this condition and is required to attain MODE 5.
B. is incorrect since RCS pressure is too high to place RHR in service (<337 psig).
C. is incorrect since the Tech Spec requires immediate action to restore one loop to operable, not 1 hour.
D. is correct as described above.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
29	Both	2	1	003000	K5.05	2.8	3.0	High

System/Evolution Name:
Reactor Coolant Pump System (RCPS)

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the RCPS:

KA Statement:
The dependency of RCS flow rates upon the number of operating RCPs

UserID:

Topic

Question Stem:

Given Unit 1 at 25% power with the following plant conditions:

- Steam Dumps are in Tave-Mode.
- Rod Control is in Manual
- 1A RCP trips

With NO operator action, what is the response of SG pressures in the OPERATING loops one minute after
the RCP tripped?

A Higher due to higher SG temperature.

B Lower due to reactor trip on LO-2 SG level.

C No change due to a constant steam demand.

D Lower due to higher steam flow.

Answer: D

Task No:

Obj No: A.HT7-09-A

Question Source: Byron exam bank

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-HT-XL-07, Ch 7, Steady State, Normal and Abnormal, Pgs 28-31

Reference:

Explanation:

This creates less total system flow; affected loop SG Tsat=Tcold from the other loops due to reverse flow resulting in essentially no steam flow from this SG, core flow is less, core delta-T rises, SGs with RCP steam more for same steam demand, thus SG pressure drops with same steam demanded and same core power produced.
A. is incorrect since just the opposite occurs.
B. is incorrect since maintaining SG level will not be a result of the RCP trip.
C. is incorrect since pressure must drop with same steam demanded from only 3 of the 4 SGs.
D. is correct as described above.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
30	Both	2	1	004000	A4.04	3.2	3.6	High

System/Evolution Name:
Chemical and Volume Control System (CVCS)

Category Statement:
Ability to manually operate and/or monitor in the control room:

KA Statement:
Calculation of boron concentration changes

UserID: Topic

Question Stem:

Unit 1 is operating at 50% power with Rod Control in Manual. A special RCS Chemistry procedure requires RAISING Tave 6°F, with no change in rod position or plant power, by changing boron concentration ONLY.

Given the following parameters:

- Initial RCS boron concentration = 600 ppm
- Moderator Temperature Coefficient = -15 pcm/°F
- Differential boron worth = -10 pcm/ppm

Which of the following is the final RCS boron concentration needed to RAISE Tave 6°F?

A 591 ppm

B 596 ppm

C 604 ppm

D 609 ppm

Answer: A

Task No:

Obj No: A.RT5-07

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reactor Theory, Ch 5, I1-RT-XL-05, Chemical Shim Control, Pgs 21-23, 30 Reference:

Explanation:

Raising temperature at constant power and a negative MTC means that negative reactivity needs to be removed (add + reactivity) to offset the change, therfore boron must be removed to raise temperature. The negative reactivity from the moderator temperature rise requested is -15 pcm/°F X 6°F = -90 pcm to offset. -90 pcm divided by -10 pcm/ppm boron worth = 9 ppm reduction in boron concentration. 600 ppm - 9 ppm = 591 ppm.

A. is correct as calculated above.

B. is incorrect and could have resulted from taking 6°F X -10 pcm/ppm, then divide by -15 pcm/°F = 4 and subtract from 600 to get 596. Misconception error.

C. is incorrect and could have happened with the misconception in B. above, then add it to 600.

D. is incorrect and could have resulted from the misconception of reactivity added being + or sign mistake and misconception.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
31	Both	2	1	004000	K4.16	2.6	3.0	Low

System/Evolution Name:
Chemical and Volume Control System (CVCS)

Category Statement:
Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following:

KA Statement:
Temperature at which the temperature control valve automatically diverts flow from the demineralizer to the VCT; reason for this diversion

UserID: Topic

Question Stem:

As CVCS Letdown temperature reaches ___(1)___, Letdown flow automatically bypasses the demineralizers to ____(2)______.

	(1)	(2)
A	133°F	protect the demin resin beads from decomposing and releasing impurities into the RCS.
B	125°F	maintain the VCT at the proper temperature for Hydrogen addition.
C	133°F	prevent reverse ion exchange from occurring resulting in RCS dilution.
D	125°F	prevent excessive channeling through the resin bed reducing ion exchange efficiency.

Answer: A

Task No:

Obj No: S.CV1-05-D

Question Source: New

Question Difficulty: Low

Time: 1

Cross Ref: BAR 1-9-E2

Reference:
GFE Fundamentals, Sect 1, Ch 7
I1-CV-XL-01, Ch 15a, CVCS, Pg 15, 16

Explanation:

CV129, Letdown Demin High Temp Divert Valve, will divert flow directly to the VCT, bypassing the demins, at 133°F. The high temperature alarm is also at 133°F on 1-9-E2. At > 140°F, the resin will overheat resulting in degradation. This may lead to release of ions exchanged as well as the resin matrix itself. These constituants are not compatible with RCS chemistry control and once activated can cause coolant activity levels to be elevated and cause pH problems.

A. is correct as described.

B. is incorrect since the actuation is at 133°F and diversion has nothing to do with H2 addition.

C. is incorrect since reverse ion exchange is not the concern. Some additonal boron is initially released as temperature rises, but this is not due to reverse ion exchange.

D. is incorrect since the setpoint is 133°F and high flow would cause channeling to occur, not high temperature.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
32	Both	2	1	005000	K5.02	3.4	3.5	Low

System/Evolution Name:
Residual Heat Removal System (RHRS)

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the RHRS:

KA Statement:
Need for adequate subcooling

UserID: Topic

Question Stem:

A LIMITATION in BOP RH-6, Operation of the RH System in Shutdown Cooling, states that "Switchover from the shutdown cooling to ECCS alignment should not be performed when the RCS Hot Leg temperature is > 260°F".

What is the reason for this LIMITATION?

A Realignment could overpressurize the RWST.

B Realignment may cause steam binding at the RH pump suction to be aligned for ECCS injection.

C Realignment may result in inadequate subcooling at the RH pump suction that is being aligned to provide shutdown cooling.

D Realignment may cause RWST temperature limit to be exceeded.

Answer: B

Task No:

Obj No: S.RH1-12

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

BOP RH-6, Operation of the RH System in Shutdown Cooling. Reference:
I1-RH-XL-01, Ch 18, Residual Heat Removal System, Pg 23

Explanation:

When an RHR train is in the SDC lineup and temperature at the suction is > 260°F and subsequently realigned for ECCS injection, the suction header change in pressure from the RCS to the RWST will cause voiding (flashing) at the pump suction or lack of subcooling for a pump start. This makes the train inoperable for ECCS injection and, until MODE 5 is entered, 1 Train is required to be in the ECCS injection lineup.
A. is incorrect since the RWST is adequately vented, overpressure would not be a concern.
B. is correct as described above.
C. is incorrect since the concern is the RHR train being realigned for ECCS injection. The train taking suction on the RCS will be well above saturation on the pump suction.
D. is incorrect since the RWST volume is so large, there would be very little temperature rise.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
33	Both	2	1	005000	K6.03	2.5	2.6	High

System/Evolution Name:
Residual Heat Removal System (RHRS)

Category Statement:
Knowledge of the effect of a loss or malfunction of the following will have on the RHRS:

KA Statement:
RHR heat exchanger

UserID:

Topic

Question Stem:

Unit 1 is shutdown with the following conditions:

- 1B Train RHR providing shutdown cooling.
- RCS pressure = 350 psig
- RCS temperature = 330°F
- RCS cooldown rate = 30°F/hour
- RHR total flow = 3300 gpm
- 1RH607, RH HX 1B Outlet Flow Control Vlv, throttled at 52% open (1500 gpm)

Flow transmitter 1FT619, 1B RH Discharge Flow, fails LOW with flow controller for 1RH619, RH HX Bypass valve, in automatic.

What will the operator observe due to this failure?

A The RCS cooldown rate will NOT change.

B The RCS cooldown rate will RISE.

C The RCS cooldown rate will DROP.

D RCS pressure would rapidly DROP.

Answer: C

Task No:

Obj No: S.RH1-11

Question Source: Byron Cert exam bank (2001) - modified

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RH-XL-01, Ch 18, Residual Heat Removal System, Pg 12, 46, 52, 59

Reference:

Explanation:

1RH619 controller is set at 3300 gpm in auto to maintain total RH pump flow at that value. The valve modulates to maintain that flow as sensed by 1FT619. With 1FT619 failing low, the 1RH619 valve will open to raise the flow sensed by 1FT619. As it opens, less RH pump discharge is through the heat exchanger and more is bypassed through the 1RH619 valve, therefore the RCS cooldown rate will drop.
A. is incorrect since less flow will be through the heat exchanger.
B. is incorrect since the cooldown rate will drop as more flow bypasses the heat exchanger.
C. is correct as described above.
D. is incorrect since the total flow will rise, but returns to the RCS, no RCS inventory is lost and a bubble exists in the PZR.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
34	Both	2	1	006000	K6.03	3.6	3.9	High

System/Evolution Name:
Emergency Core Cooling System (ECCS)

Category Statement:
Knowledge of the effect of a loss or malfunction of the following will have on the ECCS:

KA Statement:
Safety Injection Pumps

UserID: Topic

Question Stem:

Given the following plant conditions on Unit 1:

- A LOCA has occurred
- 1B SI pump tripped
- Transfer to Cold Leg Recirculation is required.
- RCS pressure is ~ 50 psig.

What is the approximate total SI pump flow indicated on the main control board and how will this value change following transfer of BOTH trains of ECCS to Cold Leg Recirculation?

	(Flow)	(Change)
A	605 gpm	lower
B	605 gpm	higher
C	1210 gpm	lower
D	1210 gpm	higher

Answer: B

Task No:

Obj No: S.EC1-03-B

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref: I1-EC-XL-01, Ch 58, ECCS, Pgs 9, 10

Reference:

Explanation:

Maximum flowrate for each SI pumps is ~ 650 gpm with 800 psig in the RCS. Below 800 psig, flow is restricted to this value. The total flow includes ~ 45 gpm recirc flow, so the actual injection will be read on the MCR indicator as ~ 605 gpm. Since the pumps create this flowrate taking suction from the RWST and creates flow based on DP across the pump, the flowrate will rise when the suction pressure rises in cold leg recirc because the RH pumps are now aligned to provide suction pressure to the SI pump. The actual suction pressure changes from ~ 20 psig to as much as 200 psig. Also, part of the alignment shift to Cold Leg Recirc closes the SI mini-flow valves, thus the 45 gpm recirc will now go into the RCS and be seen on the MCB meter. And in this case only one SI pump is running.

A. is incorrect since flow will rise due to significantly higher suction pressure to the pump.
B. is correct as described above.
C. is incorrect since only one SI pump is running and flow would be higher.
D. is incorrect since only one SI pump is running.

Date Written: 3/6/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
35	Both	2	1	007000	A2.06	2.6	2.8	High

System/Evolution Name:
Pressurizer Relief Tank/Quench Tank System (PRTS)

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Bubble formation in PZR

UserID:

Topic

Question Stem:

Unit 1 is in MODE 5 preparing to draw a bubble in the PZR with the following initial conditions in the PRT:

- PRT level = 71%
- PRT pressure = 4.5 psig
- PRT temperature = 85°F

Following venting the PZR solid:

- PRT level = 77% and stable
- PRT pressure = 7.2 psig and slowly rising
- PRT temperature = 92°F and stable

What caused this response and what is the required action?

A PZR PORV did NOT Close; CLOSE the PZR PORV Block valve.

B Gaseous Waste isolation valve did NOT close; CLOSE the PRT to GW isolation valve.

C Nitrogen Regulator has failed; CLOSE N2 Supply to PRT isolation valve.

D RCP Seal Leakoff Relief valve is lifting; VERIFY Seal Leakoff lineup to the VCT.

Answer: C

Task No:

Obj No: S.RY1-13/14/15

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:
BAR 1-12-B7, PRT PRESS HIGH

Reference:
I1-RY-XL-01, Ch 14, Pressurizer, Pgs 14, 15, 22

Explanation:

At 6 psig, RY469 auto closes to GW from the PRT. The BAR says probable cause is (1) Valve leakoff or relief valve flow, (2) PORV or Safety valve lifted, (3) Filling PRT, and (4) N2 Regulator failure. With no additional level or temperature rise, the N2 regulator would be the appropriate selection for cause. Subsequent action closes PW to PRT CNMT isolation (this would not fix the problem since there is no indication it is leaking by, i.e. no level rising) and closes N2 supply to the PRT. This would be correct action for the indicated pressure rise.
A. is incorrect because temperature and level are stable.
B. is incorrect because pressure would only be sensed from the PRT to the GW header, check valve would prevent reverse.
C. is correct by process of elimination as described above.
D. is incorrect since level is stable.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
36	Both	2	1	008000	K3.01	3.4	3.5	High

System/Evolution Name:
Component Cooling Water System (CCWS)

Category Statement:
Knowledge of the effect that a loss or malfunction of the CCWS will have on the following:

KA Statement:
Loads cooled by CCWS

UserID:

Topic

Question Stem:

Unit 1 is operating with the following conditions:

- Cooldown rate = 20°F/hour.
- 1A RH Train is in service, 1B RH Train is in standby.
- 1A RCP is running.
- 1B, 1C, and 1D RCPs are stopped.
- RCS pressure = 225 psig with a bubble in the PZR (Aux Spray valve 1CV8145 is open).
- RCS Hot Leg temperature = 230°F.
- Feeding and steaming SGs has been secured.
- Indicated letdown flow = 75 gpm.

What would be the effect of 1CC130A, Letdown HX Outlet Temperature Control Valve, failing OPEN?
(Assume no operator action and neglible change in 1A RH HX return temperature to the RCS)

A RCS temperature will RISE.

B RCS pressure will DROP.

C RCS pressure will RISE.

D Neglible effect on RCS conditions.

Answer: B

Task No:

Obj No: S.CV1-05-A, S.CV1-

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-CV-XL-01, Ch 15a, CVCS, Pg 91

Reference:

This failure would put maximum CC flow through the letdown Hx dropping letdown temperature. This would then cool the VCT.

Explanation:

Cooler water is then returned to the RCS via RCP seals, normal charging, and Aux spray to the PZR. With no operator action, RCS pressure would drop requiring manual action to restore pressure.
A. is incorrect; even if some flow is diverted from the RH Hx to provide more for the L/D Hx, this would only slow the cooldown rate. The major impact would be charging to the PZR.
B. is correct as discussed above.
C. is incorrect since overall a temperature drop would occur, thus pressure, would drop.
D. is incorrect since the colder water to the PZR would cause a significant drop in pressure.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
37	Both	2	1	010000	K1.03	3.6	3.7	High

System/Evolution Name:
Pressurizer Pressure Control System (PZR PCS)

Category Statement:
Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems:

KA Statement:
RCS

UserID: Topic

Question Stem:

Given the following conditions on Unit 1:

- Reactor power is steady at 100%.
- All systems normally aligned.
- Tave is steady on program.
- PZR level is on program and stable.
- PZR pressure is 2230 psig and begins dropping slowly.

Which of the following has occurred?

A 1LK-459, PZR Level Controller, has failed HIGH.

B 1RY456, PZR PORV, is full OPEN

C 1PT-458, PZR pressure transmitter, has failed HIGH.

D 1RY455B, PZR Spray valve, has failed to 20% OPEN.

Answer: D

Task No:

Obj No: S.RY1-25

Question Source: Byron NRC exam bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RY-XL-01, Ch 14, Pressurizer, Pgs 39, 60, 73

Reference:

Explanation:

Pressure drops from the condensing action of a large amount of spray flow with no change in PZR level.
A. is incorrect because this failure will cause both level and pressure to rise. This will call for max charging flow.
B. is incorrect since this would cause pressure to drop rapidly, not slowly.
C. is incorrect since this failure will not open a PORV. It only disables PORV reset at 2185 psig, if it were to open inadvertently.
D. is correct as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
38	Both	2	1	012000	K3.01	3.9	4.0	High

System/Evolution Name:
Reactor Protection System

Category Statement:
Knowledge of the effect that a loss or malfunction of the RPS will have on the following:

KA Statement:
CRDS

UserID: Topic

Question Stem:

Given the following Unit 1 conditions:

- Reactor power is at 100%.
- Reactor Trip Bypass breaker A (BYA) is racked in and closed for testing.
- Both Reactor Trip breakers (RTA and RTB) are closed.

What would occur if a single 15 VDC power supply failed in the 1B Train SSPS Logic cabinet?

A The redundant power supply will maintain normal SSPS Train B conditions and only a Safeguards Test Cabinet Power Failure alarm is generated.

B The reactor trips when BOTH the UV and Shunt trip coils are actuated for BYA and RTB.

C Plant conditions remain unchanged with a General Warning alarm lit for 1B train.

D The reactor trips when BOTH the UV and Shunt trip coils are actuated for RTA and RTB.

Answer: D

Task No:

Obj No: S.RP1-06/09/11

Question Source: Byron NRC exam bank (2000) - modified

Question Difficulty: High

Time: 1

Cross Ref: I1-RP-XL-01, Ch 60a, SSPS, Pgs 16-18

Reference:

Explanation:

The condition of Train A has generated a General Warning, since BYA is racked in and closed. The loss of the 15 VDC on Train B generated another General Warning. With 2 General Warnings an automatic Rx Trip is generated. An automatic Rx Trip will deenergize the UV coils on BYA, RTA, and RTB, but only energize the shunt trip coils on RTA and RTB. A manual Rx Trip actuation would also energize the shunt trip coil on BYA.
A. is incorrect since a Rx trip is generated with 2 general warnings and the alarm is activated when a 120 VAC power source to SSPS is lost.
B. is incorrect because the shunt trip coil is not actuated for BYA on an automatic Rx Trip signal.
C. is incorrect because a Rx Trip will occur since Train A already had a general warning in from BYA racked in and closed.
D. is correct as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
39	Both	2	1	013000	A4.01	4.5	4.8	High

System/Evolution Name:
Engineered Safety Features Actuation System (ESFAS)

Category Statement:
Ability to manually operate and/or monitor in the control room:

KA Statement:
ESFAS-initiated equipment which fails to actuate

UserID: Topic

Question Stem:

Given the following indications on Unit 1:

- Reactor power was at 100% when a spurious SI signal was generated.

- Reactor trip breaker B failed to OPEN.

- The spurious SI signal has cleared
- Both SI reset pushbuttons have been depressed.
- The RH pumps, SI pumps, and the 1A CV pump have been stopped.

Then a small break LOCA occurs.

What will occur as containment pressure RISES to 10 psig? (Assume no operator action is taken)

A ONLY the MSIVs and MSIV Bypass valves CLOSE.

B 1B and 1C MSIVs CLOSE, 1A and 1D MSIVs remain OPEN

C 1A RH, 1A SI, and 1A CV pumps START; all MSIVs and MSIV Bypass valves CLOSE.

D 1B RH and 1B SI pumps START; all MSIVs and MSIV Bypass valves CLOSE.

Answer: D

Task No:

Obj No: S.MS1-07-D, S.EC1-

Question Source: Byron NRC exam bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-MS-XL-01, Ch 23, Main Steam System, Pgs 10, 11
I1-EC-XL-01, Ch 58, Emergency Core Cooling System, Pg 25
I1-EF-XL-01, Ch 61, ESFAS, Pgs 17, 18

Explanation:
with the B Rx trip breaker open, B train SI reset has no seal in and therefore any additional SI actuation setpoint exceeded will cause a B train SI. A Train SI is reset and sealed in until A Rx trip breaker is cycled. There has been no change in the inputs for Main Steam Isolation Actuation. Now, when pressure rises in CNMT, B train SI will actuate at 3.4 psig and MSIVs and Bypass valves will receive a close signal at 8.2 psig (both Trains). Since offsite power is supplying the ESF buses and 1B CV pump was already running, it will continue to run, therefore only the 1B RH and 1B SI pump will start. A train ECCS will not auto start.
A. is incorrect since B train SI pumps will also start.
B. is incorrect since all MSIVs and Bypasses close and B train SI pumps start.
C. is incorrect since A train SI will not occur.
D. is correct as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
40	Both	2	1	013000	K2.01	3.6	3.8	High

System/Evolution Name:
Engineered Safety Features Actuation System (ESFAS)

Category Statement:
Knowledge of bus power supplies to the following:

KA Statement:
ESFAS/safeguards equipment control

UserID:

Topic

Question Stem:

Unit 2 Reactor trip and SI have occurred. All systems responded as designed.

1 minute into the event the following indications occur:
- 2B DG indicates tripped.
- 2B Train ESF pumps show running amps, but all RUN lights are NOT LIT.

What could have caused these indications to occur?

A SAT Feed Breaker 242-2 has tripped

B 120 VAC Bus 212 has deenergized

C 480 VAC MCC 232X has tripped

D 125 VDC Bus 212 has deenergized

Answer: D

Task No:

Obj No: S.EF1-08, S.DC1-06

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-EF-XL-01, Ch 61, ESFAS
I1-DC-XL-01, Ch 8a, 125 VDC Power Systems

Explanation:
Loss of DC power to a running DG will cause the DG to shutdown (fuel racks close), but in the control room the annunciators will indicate a trip condition. Run lights are also DC powered, as is breaker indications of position, but the amp indications are AC powered, therefore with running pumps, amps will continue to display, but run lights go out.
A. is incorrect since this would have caused the DGs to close onto their respective buses and load.
B. is incorrect since this would not cause either indication to occur.
C. is incorrect since loss of any single MCC would not have caused these indications to occur.
D. is correct as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref: none

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
41	Both	2	1	022000	A1.02	3.6	3.8	High

System/Evolution Name:
Containment Cooling System (CCS)

Category Statement:
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:

KA Statement:
Containment pressure

UserID: Topic

Question Stem:

The following conditions exist on Unit 1:

- LOCA is in progress.
- Containment Spray has actuated.
- Containment pressure is currently 18 psig.
- Containment Spray signal has been reset.
- All actions of 1BEP ES-1.3, Transfer to Cold Leg Recirculation, have been completed.

Off-site power has been lost and the DG output breakers have just CLOSED on the ESF buses.

How will the Containment Spray Pumps respond?

A Pumps will auto start 15 seconds later.

B Pumps will auto start 40 seconds later.

C If the operator immediately places BOTH CS & PHASE B ISOL switches to ACTUATE, pumps will start immediately.

D If the operator immediately places BOTH CS & PHASE B ISOL switches to ACTUATE, pumps will auto start 15 seconds later.

Answer: D

Task No:

Obj No: S.CS1-0/16

Question Source: Byron NRC exam bank (1998) - modified

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-CS-XL-01, Ch 59, Containment Spray System, Pgs 13-15

Reference:

Explanation:
CS pumps will sequence with a DG output breaker closure at 15-18 seconds or after 40 seconds depending on whether 2/4 logic is satisfied by CNMT pressure at or above 20 psig. Reset of CS & Phase B wil block auto restart of the CS pumps. For the loss of offsite power, the loads on the 4KV buses will load shed, then sequence. At 15 seconds the CS pump will restart only if the operator manually reactivates both CS & Phase B relay switches for the start logic to see an active start permissive. However, the DG sequencer will still control the actual start time of the CS pumps.
A. is incorrect since the CS pump circuit will not see HI-3 Cnmt pressure in this condition.
B. is incorrect since the CS pump circuit will not see HI-3 Cnmt pressure and would have started at 15 seconds if it had.
C. is incorrect since manual actuation is blocked by the sequencer to prevent a DG overload during sequencing.
D. is correct as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
42	Both	2	1	026000	A3.01	4.3	4.5	High

System/Evolution Name:
Containment Spray System (CSS)

Category Statement:
Ability to monitor automatic operation of the CSS, including:

KA Statement:
Pump starts and correct MOV positioning

UserID: Topic

Question Stem:

Given the following conditions on Unit 1:

- The unit is operating at 100% power.
- 1CS007A, 1A CS Pump Header Isolation valve, is OOS and CLOSED for breaker inspection.
- A spurious SI signal has actuated.
- SI is reset and ECCS has been terminated.

Five minutes later, a steamline break occurs.

- Containment pressure RISES to 25 psig.
- Main steamline pressure DROPS to 600 psig.

With no operator action, the 1A CS pump will ___(1)___ and the 1B CS pump will ___(2)___.

	(1)	(2)
A	START	NOT START
B	START	START
C	NOT START	START
D	NOT START	NOT START

Answer: B

Task No:

Obj No: S.CS1-08-C, S.CS1-09

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-CS-XL-01, Ch 59, Containment Spray System, Pgs 13-17

Reference:

Explanation:

CS pumps will auto start with 2/4 CNMT pressures at or above 20 psig (HI-3) with power to the breaker. 1CS007A/B receive an open signal, but are not interlocked to be open for a CS pump to start. The 1A CS pump will not provide any flow to CNMT with 1CS007A closed, but the pump will start as long as 1CS019A, Eductor Suction Isolation valve, from the Spray Additive Tank is open.

A. is incorrect since nothing is identified on Train B that would prevent an auto start of 1B CS pump.
B. is correct as described above.
C. is incorrect since 1CS007A closed will not prevent an auto start of 1A CS pump.
D. is incorrect since both CS pumps would receive an auto start signal and have all required interlocks satisfied to start.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
43	Both	2	1	039000	A4.01	2.9	2.8	Low

System/Evolution Name:
Main and Reheat Steam System (MRSS)

Category Statement:
Ability to manually operate and/or monitor in the control room:

KA Statement:
Main steam supply valves

UserID: Topic

Question Stem:

Unit 1 is at 3% power preparing for power ascension. Reheater Temperature Control System (RTC) is in AUTO with a Cold Start in progress.

When should OPEN indication for the 1MS009A/B/C/D, MSR MS Shutoff valves, be observed?

A Main Turbine reaches 600 rpm.

B Main Turbine load reaches 35%.

C Main Turbine reaches 1800 rpm.

D Main Turbine Generator reaches 250 Mwe.

Answer: B

Task No:

Obj No: S.MT1-08

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-MT-XL-01, Ch 35, Main Turbine and Reheaters

Reference:

Explanation:

A. is incorrect - 600 rpm is true for a Hot Start.
B. is correct - RTC in Auto and Cold Start opens valves at 35% as sensed off 1st stage HP turbine.
C. is incorrect - 1800 rpm is turbine rated speed but provides NO feedback to RTC.
D is incorrect - Mwe does not feedback to RTC.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
44	Both	2	1	039000	K5.05	2.7	3.1	Low

System/Evolution Name:
Main and Reheat Steam System (MRSS)

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the MRSS:

KA Statement:
Bases for RCS cooldown limits

UserID: Topic
Question Stem:

Unit 1 reactor has tripped. If all systems respond as expected, the 1MS009A-D, MSR MS Reheater Shutoff valves, 1MS067A-D, S/U Purge valves, and 1MS147A-D and 1MS010A-D, RHTR Temperature Control valves, will automatically CLOSE.

In 1BEP ES-0.1, Reactor Trip Response, at Step 2 the Reheater Shutoff and S/U Purge valves may need to be VERIFIED CLOSED.

What is the concern if these valves are still OPEN?

A MSR reliefs may be challenged causing internal damage to the MSRs.

B Excessive heating may occur with potential damage to the turbine LP sections during turbine coastdown.

C Excessive RCS cooldown may occur resulting in uncontrolled positive reactivity addition and subsequent challenge to RCS INTEGRITY.

D Excessive loading may be placed on the Main Condenser resulting in overpressure, thus limiting use of Steam Dumps for decay heat removal.

Answer:	Task No:		Question Source:	Question Difficulty
C	Obj No:	S.MT1-08, S.EP01-	New	Medium

Time: Cross Ref:
1

I1-MT-XL-01, Ch 35, Main Turbine and Reheaters Reference:
I1-EP-XL-01, 1BEP-0 (- ES-0.2), Reactor Trip or SI, Pg 56

Explanation:

Reactivity management is the first issue with temperature in the RCS < 557°F. These valves are addressed in the RNO for step 2 of BEP ES-0.1 to ensure excess steam demand is not causing an uncontrolled cooldown adding positive reactivity and reducing SDM. A continued excessive cooldown rate could eventually challenge RCS INTEGRITY, therefore the RNO action of ensuring these valves are closed is necessary. The valves open will continue to draw main steam through the MSR tube bundle to the condenser. This could add undue stress to the tube bundle, however, all drains should be open to the condenser. Even though conditions could be created to speculate damage to components, the primary reason for the action is the continued main steam draw creating impacts on the RCS, not what may occur to secondary components.

A. is incorrect since with proper Main Turbine Trip actuation, this pressure is dumped rapidly to the condenser.
B. is incorrect since this steam or the environment that might be heated up will be dissipated to the condenser. The LP segments are also protected from overpressure, as well as the condenser, by blowout rupture discs on the LP casings.
C. is correct as described above.
D. is incorrect since overpressure of the condenser is protected by blowout rupture discs on the LP casings.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
45	SRO	2	1	059000	A1.03	2.7	2.9	High

System/Evolution Name:
Main Feedwater (MFW) System

Category Statement:
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW System controls including:

KA Statement:
Power level restrictions for operation of MFW pumps and valves

UserID: Topic

Question Stem:

Unit 2 is operating at 95% power.

- Low Pressure Feedwater Heater Bypass Valve receives an OPEN signal due to a circuit problem.

Plant cycle efficiency will ________________________________ as a result of this valve opening.

A lower, driving Reactor Power up, since more energy from the reactor is required due to feedwater entering the SGs at a lower temperature

B rise, driving reactor power down, due to less work required to pump the water through the low pressure feedwater heaters

C lower, driving Reactor Power up, since mass flowrate entering the SGs will be higher

D rise, driving Reactor Power down, due to the rise in feedwater temperature resulting in raising the efficiency of the high pressure feedwater heaters

Answer: A

Task No:

Obj No: A.HT7-02

Question Source: Millstone NRC exam bank (1997)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference: I1-HT-XL-07, Ch 7, Steady State Operation, Normal and Abnormal, Pgs 8, 14

Explanation:

This actuation results in a drop in feedwater reheating for the return feed to the steam generators. Efficiency is basically Work of the turbine less work of the FW, CD/CB pumps divided by Q reactor. Overall, for a constant load on the main turbine, the Net work will be relatively unchanged, however, the Q of the reactor will have to be higher with the lower feedwater temperature to maintain the same turbine work. This will result in lowering plant efficiency.
A. is correct as described above.
B. is incorrect since efficiency is net work divided by Q reactor. The Net work change is minimal and the steam flow through the turbine to maintain MWs out will have to rise, since Steam pressure will drop slightly since Tave will drop with no operator action to maintain Tave the same.
C. is incorrect since mass flowrate change in the secondary will have much less effect than the Q reactor change assuming MWs out remain unchanged.
D. is incorrect since just the opposite will occur as described above.

Date Written: 3/7/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
46	Both	2	1	061000	2.1.28	3.2	3.3	Low

System/Evolution Name:
Auxiliary / Emergency Feedwater (AFW) System

Category Statement:
Conduct of Operations

KA Statement:
Knowledge of the purpose and function of major system components and controls.

UserID:

Topic

Question Stem:

The 1B AF pump can be started locally at the 364' level in the Aux Building near the CC Heat Exchangers.

One of the switch positions is "START WITH BYPASS".

Which of the following trip signals does this switch position bypass?

A All trips are bypassed.

B Low lube oil pressure.

C Low suction pressure.

D High Jacket Water temperature.

Answer: C

Task No:

Obj No: S.AF1-05/15

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-AF-XL-01, Ch 26, Auxiliary Feedwater System, Pg 10

Reference:

Explanation:

This Switch position can only be used if there is a Lo-Lo suction trip and a fire is present in the AEER or Aux building elevation 383', 401' or 426'. The switch also has a NORMAL position that allows starts from normal locations and a START position to allow a remote start with suction pressure > Lo-Lo suction trip.
A. is incorrect since it will bypass the Lo-Lo suction trip only. The other trips are still in the circuit but not as likely impacted by a fire.
B. is incorrect since this is never bypassed (except a time delay on start for pressure to buildup).
C. is correct as described above.
D. is incorrect since this is never bypassed.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
47	Both	2	1	061000	K4.08	2.7	2.9	Low

System/Evolution Name:
Auxiliary / Emergency Feedwater (AFW) System

Category Statement:
Knowledge of AFW System design feature(s) and/or interlock(s) which provide for the following:

KA Statement:
AFW recirculation

UserID: Topic

Question Stem:

Which of the following describes how Auxiliary Feedwater Pumps are protected from damage when running with feed flow to the SGs isolated?

A Recirculation is controlled at ~100 gpm back to the suction source by an air operated control valve that senses downstream flowrate.

B Recirculation is limited to ~100 gpm back to the suction source by an in-line orifice and is isolated to the CST with an air operated isolation valve that fails open on loss of air pressure.

C At least 85 gpm recirculation flow is maintained by a manual throttle valve with a motor operated isolation valve that automatically closes when the SX suction valves open.

D A maximum recirculation flow of 85 gpm is controlled by an in-line orifice with an air operated isolation valve to the CST that automatically closes when the SX recirculation valve opens.

Answer: B

Task No:

Obj No: S.AF1-14

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-AF-XL-01, Ch 26, Auxiliary Feedwater System

Reference:

Explanation:

A is incorrect because flow is not regulated by a control valve.
B. is correct because this is how the sytem is designed.
C. is incorrect because 100 gpm is the design to ensure at least 85 gpm is met and an orifice is used with an AOV for isolation.
D. is incorrect because 100 gpm is the design to ensure at least 85 gpm is met and the AOV closes if BOTH respective SX valves are NOT fully closed.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
48	Both	2	1	062000	K2.01	3.3	3.4	Low

System/Evolution Name:
A.C. Electrical Distribution System

Category Statement:
Knowledge of bus power supplies to the following:

KA Statement:
Major system loads

UserID: Topic

Question Stem:

A reactor trip has just occurred on Unit 1. The Automatic Bus Transfer (ABT) failed to operate for Bus 157.

Which of the following loads is UNAVAILABLE?

A 1A Main Feed pump

B 1B Reactor Coolant pump

C 1C Heater Drain pump

D 1D Condensate/Condensate Booster pump

Answer: C

Task No:

Obj No: S.AP1-12-A

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Low

Time: 1

Cross Ref:

I1-AP-XL-01, Ch 4, AC Electrical Power Systems, Pg 34

Reference:

Bus 157 is a 6.9KV bus that provides breakers to 1A RCP, 1A and 1C HD pumps, and the RSH transformers. Explanation:
A. is incorrect because it's power source is 6.9KV bus 156.
B. is incorrect because it's power source is 6.9KV bus 156.
C is correct
D is incorrect because it's power by 6.9KV bus 158

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
49	Both	2	1	063000	K1.02	2.7	3.2	High

System/Evolution Name:
D.C. Electrical Distribution System

Category Statement:
Knowledge of the physical connections and/or cause-effect relationships between the D.C. Electrical System and the following systems:

KA Statement:
AC electrical system

UserID:

Topic

Question Stem:

Unit 2 was at 100% power in a normal at power lineup.

- Power is lost to DC Bus 212
- 5 minutes later, SAT 242-1 develops a fault

What is the status of ESF Bus 242 immediately following the SAT fault?

A Energized from 2B DG.

B Energized from SAT 242-2.

C Deenergized with all feed breakers tripped.

D Deenergized with ACB 2422 closed.

Answer: D

Task No:

Obj No: S.DC1-09

Question Source: Byron NRC exam bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-DC-XL-01, Ch 8a, 125 VDC Power Systems, pgs 19-20

Reference:

Explanation:

A. is incorrect since 2B DG will not start due to no DC power.
B. is incorrect since SAT 242-2 will also be lost from the tripping of th SAT's feed breaker from the switchyard.
C. is incorrect since all feed breakers will still be closed without DC to open them.
D. is correct since there will be no power source to the bus and 2422 will not have opened since DC was lost.

Date Written: 3/8/2006

Author: M. Jorgensen

App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
50	Both	2	1	064000	2.1.2	3.0	4.0	Low

System/Evolution Name:
Emergency Diesel Generator (ED/G) System

Category Statement:
Conduct of Operations

KA Statement:
Knowledge of operator responsibilities during all modes of plant operation.

UserID:

Topic

Question Stem:

The crew is performing 1BEP ES-1.2, Post LOCA Cooldown and Depressurization.

- ESF busses are supplied by the diesel generators (DG).
- The DGs have been continuously loaded to 6000 KW at 1050 amps for 1 hour.

By design, how much longer can the DGs remain running at this present load?

A DGs must be secured immediately.

B 1 hour.

C 1999 hours.

D Indefinitely.

Answer: B

Task No:

Obj No: S.DG1-01

Question Source: Byron NRC exam bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-DG-XL-01, Ch 9, Diesel Generator and Auxiliary System, Pg 32

Reference:

Explanation:

The DGs are rated for 5500 Kw @954 amps continuous; 5500-5935 Kw @1030 amps for 2000 hours; 5935-6050 Kw @1050 amps for 2 hours.
A. is incorrect since this load is allowed for 2 hours.
B. is correct as stated above.
C. is incorrect since this load is allowed for 2 hours.
D. is incorrect since this load is allowed for 2 hours.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
51	Both	2	1	064000	K4.10	3.5	4.0	Low

System/Evolution Name:
Emergency Diesel Generator (ED/G) System

Category Statement:
Knowledge of ED/G System design feature(s) and/or interlock(s) which provide for the following:

KA Statement:
Automatic load sequencer: blackout

UserID: Topic

Question Stem:

Given the following conditions on Unit 2:

- The reactor is at 100% power.
- A grid problem has just LOWERED voltage to 3700 volts on Buses 241 and 242.

With no operator action and voltage sustained at 3700 volts, what is the SEQUENCE of events for this condition?

A 2A and 2B DGs start after ~310 seconds, then both ESF buses deenergize.

B 2A and 2B DGs start immediately, then both ESF buses deenergize.

C Both ESF buses deenergize after ~310 seconds, then 2A and 2B DGs start.

D Both ESF buses deenergize immediately, then 2A and 2B DGs start.

Answer: C

Task No:

Obj No: S.AP1-10-A

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-AP-XL-01, Ch 4, AC Electrical Power System, Pg 40

Reference:

Explanation:

With voltage degraded to < 3847.5 volts but > 2870 volts, a 310 second time delay is actuated and must time out to cause the feeder breakers for the ESF buses to trip. Once tripped, the DGs see an UV and receive a start signal.
A. is incorrect because the sequence is not correct.
B. is incorrect because both the time frame and sequence are not correct.
C. is correct
D. is incorrect because the timer must time out first.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
52	Both	2	1	073000	A2.02	2.7	3.2	High

System/Evolution Name:
Process Radiation Monitoring (PRM) System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Detector failure

UserID: Topic

Question Stem:

A liquid release is in progress from Release Tank 0WX01T.

- RM-11 alarm is received and acknowledged to be 0PR01J, Liquid Radwaste Effluent Rad Monitor, with Color Code: BLUE.

What is the impact of this alarm and appropriate action(s), if any?

A Detector RM-80 micro-processor has lost communication with the RM-11; VERIFY with RW operator that release flow rate has NOT changed; NOTIFY Chemistry to sample now and each 4 hours until communications is reestablished.

B Detector may have failed; VERIFY with RW operator that release flow rate has NOT raised; NOTIFY Chemistry to recalculate and verify release rate and continue to sample each 4 hours until the detector is restored.

C OPERATE FAILURE is indicated; VERIFY with RW operator that the release has TERMINATED; NOTIFY Chemistry to calculate new release limits; initiate LCOAR for 0PR01J and reestablish conditions for a new release.

D OPERATE FAILURE is indicated; VERIFY 0PR10J, Station Blowdown monitor, is in service and below the ALERT limit; If YES, the release may continue; initiate LCOAR for 0PR01J.

Answer: C

Task No:

Obj No: S.AR1-04-B-01/09/18,

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-AR-XL-01, Ch 49, Radiation Monitors
I1-WX-XL-01, Ch 48, Liquid Radwaste
RM-11 BAR for 0PR01J
BCP-400-TWX01, Liquid Radwaste Release Form For Realease Tank 0WX01T

Explanation:
The BLUE Code on the RM-11 means an OPERATE FAILURE has occurred, which could be several things from detector failure to loss of sample flow, but the exact cause will not be readily apparent at the RM-11. Since this is the case, the output will be generated to any auto actuation components as if the monitor is in a HIGH alarm condition. For 0PR01J, this will close the liquid release isolation valve, stopping the release. This is the BAR action for this condition. This detector INOPERABLE requires Tech Spec/TRM action, thus LOCAR initiation. Release may be reestablished and resumed without this monitor using alternate procedure steps.
A. is incorrect because this Color Code is Magenta and does not constitute action for release effects. 0PR01J is still operating and OPERABLE.
B. is incorrect, this reason is plausible, but isolation of the release path is expected to occur.
C. is correct per the RM-11 BAR
D. is incorrect, this reason is plausible, but isolation of the release path is expected to occur.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
53	Both	2	1	076000	A3.02	3.7	3.7	High

System/Evolution Name:
Service Water System (SWS)

Category Statement:
Ability to monitor automatic operation of the SWS, including:

KA Statement:
Emergency heat loads

UserID: Topic

Question Stem:

Unit 1 was being synchronized to the grid when a steamline break occurred in containment

- 2 minutes later, a switchyard fault caused both Unit 1 SATs to deenergize.
- Containment pressure peaked at 16.5 psig.

When would the 1A SX pump re-start?

A 5 seconds after the 1A CS pump.

B Between the start of 1A CV pump and 1A RH pump.

C 10 seconds before 1A AF pump.

D Coincident with the start of 1A and 1C RCFCs.

Answer: C

Task No:

Obj No: S.DG1-07-C, S.SX1-

Question Source: Byron NRC exam bank (1998) - modified

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-DG-XL-01, Ch 9, Diesel Generators and Auxiliaries, pgs 36-38
I1-SX-XL-01, Ch 20, Essential Service Water System, pgs 48-49

Explanation:
1A SX pump will always sequence on the DG at 25 seconds. The 1A CS pump sequences at 15-18 seconds or after 40 seconds if an auto actuation is present (at or above 20 psig in containment). The SI pump starts between the CV and RH pump. The RCFC start at time 0 in the sequence. Regardless of whether an SI occurred or just loss of offsite power the SX pump always starts 10 seconds before the 1A AF pump, which will auto sequence at the same time for either event.
A. is incorrect because the 1A CC pump would be 5 seconds after 1A CS may have received it's 1st start permissive.
B. is incorrect since the 1A SI pump would sequence at 10 seconds which is between these two pumps.
C. is correct since the 1A AF pump would start at 35 seconds and the 1A SX pump always sequences at 25 seconds.
D. is incorrect because the RCFCs will start at time 0 in this sequence being supplied by 480 VAC MCCs that are powered as soon as the DG output breaker closes.

Date Written: 3/8/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
54	Both	2	1	078000	K3.02	3.4	3.6	High

System/Evolution Name:
Instrument Air System (IAS)

Category Statement:
Knowledge of the effect that a loss or malfunction of the IAS will have on the following:

KA Statement:
Systems having pneumatic valves and controls

UserID: Topic

Question Stem:

Unit 2 is at 100% power with all systems in normal lineup, when Instrument air is lost to the containment.

Complete the following statement.

With no operator action for the next 30 minutes, Pressurizer (PZR) pressure will RISE__________.

A until the PZR PORVs will cycle to control pressure.

B initially, but will control at 2235 psig due to PZR Sprays opening.

C causing backup heaters to deenergize with the variable heaters remaining full on.

D continuously to the PZR pressure Reactor trip setpoint due to loss of letdown and the rise in charging flow.

Answer: A

Task No:

Obj No: S.RY1-05/25A/B,

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-RY-XL-01, Ch 14, Pressurizer, Pgs 33
I1-CV-XL-01, Ch 15a, CVCS, Pgs 47
1/2BOA SEC-4, Loss of IA, steps 4, 5 and Table A

Explanation:
This is an integrated plant question. CV letdown and normal charging fail closed. The seal injection flow is maintained resulting in a net rise in RCS inventory. This causes PZR level to rise, thus PZR pressure will rise as the bubble is compressed. The loss of air has caused the normal PZR spray valves to fail closed, Aux spray is isolated, thus pressure will rise to the PORV's lift setpoint and lift because the PORVs have a reservoir (accumulator) that maintains operation of the PORVs with no IA available.
A. is correct as described above.
B. is incorrect since the normal spray valves require IA to open.
C. is incorrect since the insurge will cause B/U heaters to energize, the variable heaters will be off with the higher pressure.
D. is incorrect since PORV's will still function to keep pressure below the reactor trip setpoint.

Date Written: 3/8/2006 Author: M.Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
55	Both	2	1	103000	A2.03	3.5	3.8	High

System/Evolution Name:
Containment System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the Containment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Phase A and B isolation

UserID: Topic

Question Stem:

A LOCA has occurred on Unit 1.
- Immediate actions in 1BEP-0, Reactor Trip or Safety Injection, are complete
- Containment pressure is 24 psig

The following Group Monitor lights are NOT LIT:

- 1CV8152, Letdown Line Containment Isolation Valve
- 1CC685, CC From RCPs Thermal Barrier Isolation Valve
- 1CS019B, Eductor 1B Spray Add Valve

What are the FIRST actions required in 1BEP-0, Reactor Trip or Safety Injection, for this indication?

A Manually actuate Phase A, CS & Phase B.

B VERIFY position of each valve; Manually position each valve using the control switch as required.

C Manually actuate SI, CS & Phase B.

D VERIFY associated opposite Train valve CLOSED, then continue in 1BEP-0.

Answer: A

Task No:

Obj No: T.EP01-05

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:
1BEP-0, Reactor Trip or SI, Steps 8 and 14

Reference:
I1-EP-XL-01, 1BEP-0, Reactor Trip or SI, pgs 12, 14

Explanation:
BEP-0 RNO action for each of these group lights NOT LIT specifically states to Manually Actuate Phase A for 1CV8152, and Manually Actuate CS&Phase B (2 of 2 switches) for 1CC685 and/or 1CS019B.
A. is correct as stated in the procedure.
B. is incorrect since this is followup action if the Manual actuations were not successful.
C. is incorrect since SI has already been manually actuated by procedure as a backup to auto actuation early in the event.
D. is incorrect since this is also a followup action after Manual actuation has been attempted.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
56	Both	2	2	001000	K2.01	3.5	3.6	High

System/Evolution Name:
Control Rod Drive System

Category Statement:
Knowledge of bus power supplies to the following:

KA Statement:
One-line diagram of power supply to M/G sets

UserID: Topic

Question Stem:

Unit 1 is operating at 50% power in a normal at power lineup.

Which of the following would cause Alarm 1-10-D8, ROD DRIVE M/G SET TROUBLE, to alarm?

A Loss of 4KV Bus 141

B Loss of 4KV Bus 142

C Loss of 4KV Bus 143

D Loss of 120 VAC Bus 113

Answer: C

Task No:

Obj No: S.RD1-11-A

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

BAR 1-10-D8, Rod Drive M/G Set Trouble Reference:
I1-RD-XL-01, Ch 28, Rod Control System, Pgs 20, 53

Explanation:

The power supplies to the MG sets are 480 VAC MCCs 133Y and 134Y, which are fed by 4KV buses 143 and 144. If power is lost to a running MG set with the other MG set running, the breakers for the deenergized MG set will trip on reverse power, which is one of the inputs to the alarm. Other alarm inputs are associated with other running breaker faults (i.e. OC, OV, Grnd).
A. is incorrect since this is an ESF power supply that could be x-tied to supply bus 143, but that is NOT a normal lineup.
B. is incorrect since this is an ESF power supply that could be x-tied to supply bus 144, but that is NOT a normal lineup.
C. is correct since this would cause 1A RD MG set to lose power (i.e. 133Y will deenergize) causing a reverse power trip from the other energized MG set. This will cause the alarm.
D. is incorrect since there is NO interface or impact to the MG sets from a loss of this bus.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
57	Both	2	2	002000	2.1.32	3.4	3.8	Low

System/Evolution Name:
Reactor Coolant System (RCS)

Category Statement:
Conduct of Operations

KA Statement:
Ability to explain and apply all system limits and precautions.

UserID: Topic

Question Stem:

Unit 1 is in MODE 4 during a plant heatup with the following conditions:

- RCS temperature = 300°F
- RCS pressure = 400 psig
- PZR level = 33%
- Preparations are in progress to start the FIRST RCP.

Complete the following statement of applicability.

Per BOP RC-1, Startup of a Reactor Coolant Pump, the requirement of having < 50°F difference between S/G temperature and the associated RCS loop temperature_____________

A is NOT applicable since this is the first RCP to be started.

B ensures RCP seal parameters remain within normal operating range.

C prevents an overpressure event in the RCS.

D provides adequate NPSH at the suction of the selected RCP.

Answer: C

Task No:

Obj No: S.RC2-09-C

Question Source: Byron NRC exam bank (1998)

Question Difficulty: Low

Time: 1

Cross Ref:

I1-RC-XL-02, Ch 13, Reactor Coolant Pump, Pg 31
Tech Spec 3.4.6 and 3.4.7

Reference:

Explanation:

This is the Tech Spec requirement, thus preventing a challenge to LTOP, due to the rapid distribution of the higher energy fluid when an RCP is initially started. The limit is the analyzed range boundary.

A. is incorrect since this is a Tech Spec limit and it always applies.
B. is incorrect since the seal parameters are not impacted by this requirement.
C. is correct as part of the LTOP analysis to prevent overpressurization of the RCS on an RCP start.
D. is incorrect since this is established strictly by the P/T limits in th RCS alone.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
58	Both	2	2	011000	K3.03	3.2	3.7	High

System/Evolution Name:
Pressurizer Level Control System (PZR LCS)

Category Statement:
Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following:

KA Statement:
PZR PCS

UserID: Topic

Question Stem:

Unit 1 is at 50% power with all systems in Auto, except Rod Control which is in Manual.
- Letdown flow is 75 gpm.
- RCS Loop 1C Tave channel fails HIGH.

What is the response of PZR Pressure Control AFTER 5 minutes with no operator action?

A One PORV is cycling, Sprays FULL OPEN.

B Spray valves CLOSED, Backup heaters ON, Variable heaters full ON.

C Spray valves Throttled OPEN, Backup heaters OFF, Variable heaters OFF.

D Spray valves Throttled OPEN, Backup heaters ON, Variable heaters OFF.

Answer: C

Task No:

Obj No: S.RY1-21I

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RY-XL-01, Ch 14, Pressurizer, Pgs 25, 26, 28

Reference:

Explanation:

This failure inputs to PZR level Control Program level setpoint now calling for 100% power PZR level. Charging flow rate will immediately rise, raising PZR level. As PZR level rises, pressure rises causing variable heaters to go to minimum (OFF) and sprays to open to maintain pressure control. Sprays have more than enough capacity to prevent reaching the PORV lift setpoint on this transient. Backup heaters should already be OFF and remain OFF. They will reenergize on a > 5% level deviation above program normally expected on an insurge, but in this case the deviation is the other direction and they won't reenergize.
A is incorrect because the sprays will prevent reaching the PORV setpoint.
B. is incorrect because this response would be to a drop in PZR level.
C. is correct as described above.
D. is incorrect because sprays will be on, variable heaters will be off, but backup heaters will be off also.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
59	Both	2	2	014000	K4.03	3.2	3.4	Low

System/Evolution Name:
Rod Position Indication System (RPIS)

Category Statement:
Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following:

KA Statement:
Rod bottom lights

UserID: Topic

Question Stem:

Which of the following will cause the Rod Bottom LED to FLASH?

A DRPI General Warning

B DRPI Data A OR B failure

C An Ejected rod

D A Dropped rod from 9 steps during withdrawal

Answer: C

Task No:

Obj No: S.PI1-04/06

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-PI-XL-01, Ch 29, Pgs 9-11, 18Reference:

Several failures will cause the Rod Bottom lights to flash including DRPI Data A AND Data B failures coincident with one another,
Explanation:
Data A and B differ by more than 1 bit, sum of Data A and B exceeds 38 bits, and an ejected rod.
A. is incorrect since this is caused by Data A or Data B failure, but not both at the same time or an urgent alarm is present for one particular rod or rods.
B. is incorrect since this brings in a DRPI General Warning.
C. is correct as described above.
D. is incorrect since this would bring in the light solid.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
60	Both	2	2	016000	K1.12	3.5	3.5	High

System/Evolution Name:
Non-Nuclear Instrumentation System (NNIS)

Category Statement:
Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems:

KA Statement:
S/G

UserID: Topic

Question Stem:

Unit 2 is at 35% power with all systems in normal lineup.

What failure will cause an INITIAL DROP in feedwater flow to ALL SGs?

A PT-505, Turbine First Stage Impulse Pressure, fails LOW.

B PT-506, Turbine First Stage Impulse Pressure, fails LOW

C PT-507, Main Steamline Pressure, fails LOW.

D PT-508, Main Feedwater Header Pressure, fails LOW.

Answer: C

Task No:

Obj No: S.FW2-16

Question Source: Byron NRC exam bank (1998)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-FW-XL-01, Ch 27, SG Water Level Control System, Pg 20

Reference:

Explanation:

PT-506 is input to FWP turbine speed control for maintaining delta-P program across the FRVs. A low failure would denote a large delta-P and the FWP speed will slow down to lower the delta-P, thus reducing FW flow to the SGs. The same effect would occur if PT-508 failed high.
A. is incorrect since this PT does not input to the SGWLC program.
B. is incorrect since this PT does not input to the SGWLC program.
C. is correct as described above.
D. is incorrect since this input will cause the SG's level to go up, a failure high would produce the same response as B.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
61	Both	2	2	017000	K5.01	3.1	3.9	Low

System/Evolution Name:
In-Core Temperature Monitor (ITM) System

Category Statement:
Knowledge of the operational implications of the following concepts as they apply to the ITM System:

KA Statement:
Temperature at which cladding and fuel melt

UserID: Topic

Question Stem:

Implementation of 2BFR C.1 is safety significant, when CETCs are > 1200°F, because additional operator action is required to ___________.

A Prevent core uncovery.

B Provide core cooling to stop the hydrogen generation from the zircaloy-water reaction.

C Limit containment pressure to less than the design pressure.

D Provide core cooling to prevent exceeding peak clad temperature limit.

Answer: D

Task No:

Obj No: S.FR02-01

Question Source: Byron exam bank

Question Difficulty: Medium

Time: 1

Cross Ref: I1-FR-XL-02, BFR-C.1, C.2, C.3, Pg 2, 3

Reference:

This is the express purpose of the BFR C-series, protect the first barrier (clad). Explanation:
A. is incorrect because the fact is the core uncovery has already occurred at this temperature.
B. is incorrect because this is a by-product of clad degradation and a process of embrittlement. Clad breach is the concern.
C. is incorrect because this barrier is addressed in BFR Z-series.
D. is correct as described above.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
62	Both	2	2	029000	A3.01	3.8	4.0	Low

System/Evolution Name:
Containment Purge System (CPS)

Category Statement:
Ability to monitor automatic operation of the Containment Purge System, including:

KA Statement:
CPS isolation

UserID:

Topic

Question Stem:

Which Containment Radiation Monitor provides a signal to automatically actuate a Containment Vent Isolation?

A AR011, Containment Fuel Handling Incident monitor.

B PR011, Containment Atmosphere monitor.

C AR014, Containment General Area monitor.

D PR001, Containment Purge Effluent monitor.

Answer: A

Task No:

Obj No: S.AR1-04-A-02

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-AR-XL-01, Ch 49, Radiation Monitors, Pg 30

Reference:

Explanation:

AR011 and AR012 respectively isolate Train A and Train B Primary (Not used term at Byron) CNMT (Vent at Byron) Purge (VQ) valves.
A. is correct as described above.
B. is incorrect because this monitor in alarm isolates the CNMT Air Sampling Panel and monitor.
C. is incorrect- this monitor provides indication and alarm only.
D. is incorrect- this monitor provides indication and alarm only.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
63	Both	2	2	034000	K6.02	2.6	3.3	Low

System/Evolution Name:
Fuel Handling Equipment System (FHES)

Category Statement:
Knowledge of the effect of a loss or malfunction of the following will have on the Fuel Handling System:

KA Statement:
Radiation monitoring systems

UserID: Topic

Question Stem:

While using the Spent Fuel Pool Crane to move new fuel into the Spent Fuel Pool, radiation monitor 0RE-AR039, Fuel Handling Building Crane Monitor, alarms HIGH.

What ACTION for the Fuel Handling Building Crane is affected?

A Traverse of the bridge and trolley.

B Both lowering and raising the hoist.

C Both Traverse of the trolley and raising the hoist.

D Raising the hoist only.

Answer: D

Task No:

Obj No: S.AR1-04-A-03

Question Source: Byron NRC exam bank (1998)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-AR-XL-01, Ch 49, Radiation Monitors, Pg 27

Reference:

Raising the hoist is the only movement that is inhibited. Explanation:
A. is incorrect since this movement is not inhibited.
B. is incorrect since lowering the hoist is not inhibited.
C. is incorrect since trolley movement is not inhibited.
D. is correct as stated above.

Date Written: 3/9/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
64	Both	2	2	041000	A1.02	3.1	3.2	High

System/Evolution Name:
Steam Dump System (SDS) and Turbine Bypass Control

Category Statement:
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including:

KA Statement:
Steam pressure

UserID: Topic

Question Stem:

During a Unit 2 plant cooldown, the following conditions exist:

- RCS loop Tave:
  - 550°F Loop 1 is lowering
  - 548°F Loop 2 is lowering
  - 551°F Loop 3 is lowering
  - 548°F Loop 4 is lowering

- Steam header pressure is 1030 psig and lowering.
- Steam Dump Mode Selector switch is in STM PRESS MODE.
- Steam Dump Controller is in MAN, set at 30% demand.

The operator momentarily places the Train A and Train B Steam Dump Bypass Interlock switches to BYPASS and then releases them.

What is the Steam Dump valve status following this action?

A All valves are fully CLOSED.

B Three valves in groups 1, 2, and 3 will OPEN.(9 total)

C Three valves in group 1 only will OPEN.

D One valve in groups 1, 2, and 3 will fully OPEN.(3 total)

Answer: C

Task No:

Obj No: S.DU1-04-C/07-B

Question Source: Byron NRC exam bank (2000)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-DU-XL-01, Ch 24, Steam Dumps, Pg 12 Reference:

Explanation:

If temperature on 2/4 Tave channels is below 550°F, all valves close and when BYPASS INTERLOCK is selected on both trains, the steam dumps will reopen on demand, however, only group 1 valves can open below 550°F, without jumpering the control circuit.
A is incorrect since this action will allow group 1 valves to open with a demand on the controller.
B. is incorrect since groups 2 and 3 valves are blocked from opening below 550°F.
C. is correct as described above.
D. is incorrect since 3 total valves is correct, but they are all in group 1; groups 2 and 3 are blocked below 550°F.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
65	Both	2	2	071000	A4.24	2.9	3.4	Low

System/Evolution Name:
Waste Gas Disposal System (WGDS)

Category Statement:
Ability to manually operate and/or monitor in the control room:

KA Statement:
The double verification required before waste gas release

UserID: Topic

Question Stem:

While preparing to perform BCP 400-TWASTE GAS, Gaseous Effluent Release Form: Waste Gas Decay Tank, you discover that 0PR02J, Gas Decay Tank Effluent, Radiation Monitor is INOPERABLE.

It is still desired to perform the Gaseous Release.

Which of the following does NOT REQUIRE an independent verification for the release to proceed?

A Lifting Leads BOP GW-13, 0PR02J Interlock Function Defeat.

B Placing the Gas Decay Tank, that will be released, in Storage alignment per BOP GW-6, Realignment of Gas Decay Tanks.

C Sampling the Gas Decay Tank that will be released for activity.

D Calculating the release rate of the Gas Decay Tank that will be released.

Answer: B

Task No:

Obj No: S.GW1-12

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:

BCP 400-TWASTE GAS, Gaseous Effluent Release Form: Waste Gas Decay Tank.
BOP GW-13, 0PR02J Interlock Defeat Function.
BOP GW-6, Realignment of Gas Decay Tanks
I1-GW-XL-01, Ch 46, Gaseous Radwaste System. Pgs 14, 20, 21
TRM 3.11.b, Radioactive Gaseous Effluent Monitoring Instrumentation

Explanation:

BCP 400-TWASTE GAS requires an independent verification (also TRM 3.11.b) of tank activity and release rate. Also the procedure requires BOP GW-13 be performed to defeat the interlock function of the gas release islolation valve, 0GW014. This involves lifting leads and always requires an independent verification. The realignment of the Gas Decay tanks is a normal function for operations and does not require an independent verification.
A. is incorrect since this involves lifting leadss, which always reqires IV.
B. is correct since this is a normal operator evolution and does NOT require IV.
C. is incorrect since the procedure and the TRM requires this action for 0PR02J inoperable.
D. is incorrect since the procedure and the TRM requires this action for 0PR02J inoperable.

Date Written: 3/16/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
66	Both	3		194001	2.1.12	2.9	4.0	High

System/Evolution Name:
Generic

Category Statement:
Conduct of Operations

KA Statement:
Ability to apply technical specifications for a system.

UserID:

Topic

Question Stem:

Unit 2 is in MODE 1. Due to a turbine malfunction and a trip of the running containment chiller, the following conditions exist:

- Current time = 1000.
- RCS Tave = 549°F.
- PZR pressure = 2202 psig.
- Containment pressure = 1.1 psig.
- Containment temperature (Ave of running RCFCs) = 122°F

Which Technical Specification LCO parameter MUST BE RESTORED by 1030 to allow continued operation in MODE 1? (Assume turbine problem has been corrected)

A RCS Tave.

B PZR pressure.

C Containment pressure.

D Containment temperature.

Answer: A

Task No:

Obj No: S.RC1-12, S.PC1-08

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref: Tech Specs 3.4.1, 3.4.2, 3.6.4 and 3.6.5

Reference:
I1-RC-XL-01, Ch 12, Reactor Coolant System, Pg 41
I1-PC-XL-01, Ch 40, Primary Containment, Pgs 25, 26

Explanation:
RCS Tave is required to be at or above 550°F when critical per Tech Spec 3.4.2.
A. is correct as stated above.
B. is incorrect since PZR pressure is required to be restored at or above 2209 psig, per Tech Spec 3.4.1, within 2 hours.
C. is incorrect since CNMT Pressure is required to be within limits (-.1 to +1.0 psig) within 1 hour per Tech Spec 3.6.4.
D is incorrect since CNMT Temperature is required to be at or below 120°F within 8 hours per Tech Spec 3.6.5.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
67	Both	3		194001	2.1.27	2.8	2.9	Low

System/Evolution Name: Generic

Category Statement: Conduct of Operations

KA Statement:
Knowledge of system purpose and or function.

UserID: Topic

Question Stem:

In accordance with the Operating Department Standards found in BAP 300-1, Byron Addendum to Conduct of Operations Manual, which of the following would meet the "Operator's judgement" call for taking a controller from AUTO to MANUAL?

A To establish the control output to desired setpoint faster during a load ramp.

B Periodically to check if MANUAL is tracking AUTO.

C Periodically to RESET the integral during a power ramp.

D Automatic response is NOT consistent with changing plant conditions.

Answer: D

Task No:

Obj No: T.AM03-29

Question Source: New

Question Difficulty: Low

Time: 1

Cross Ref:
BAP 300-1, OP-AA-100, Conduct of Operations Manual, Byron Addendum

Reference:
I1-AM-XL-70, BAP 300-1, OP-AA-101-101, Conduct of Operations Maunual, Byron Addendum, Pg

Explanation:
The statement in BAP 300-1, C.1.h. says " the operator may place a controller in the manual mode from the automatic mode whenever, in the operator's judgement, continued automatic operation is unsafe or whenever it may cause any unnecessary transients. This should only be done when conditions are "stable and under control", or when it is apparent that continued operation would aggravate or worsen the plant conditions".
A. is incorrect since it does not meet the intent of this standard.
B. is incorrect since it does not meet the intent of this standard.
C. is incorrect since it does not meet the intent of this standard.
D. is correct as described above.

Date Written: 3/16/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
68	Both	3		194001	2.2.22	3.4	4.1	Low

System/Evolution Name:
Generic

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID: Topic

Question Stem:

An overpressure event caused the Technical Specification Safety Limit for RCS Pressure to be exceeded in MODE 1 at 10:00, by what time must the Unit be in HOT STANDBY with RCS Pressure within limits?

A 10:05

B 10:15

C 10:30

D 11:00

Answer: D

Task No:

Obj No: S.RC1-12

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-RC-XL-01, Ch 12, Reactor Coolant System, Pg 35

Reference:
Tech Spec 2.1.2

Tech Spec 2.1.2 actions in MODE 1or 2 require reducing pressure to 2735 psig or below and being in MODE 3 within 1 hour.

Explanation:
A. is incorrect since this is the time required to restore pressure to within limits if the Unit is in MODEs 3, 4, or 5.
B. is incorrect because although plauseable, not correct.
C. is incorrect because although plauseable, not correct.
D. is correct as stated above.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
69	Both	3		194001	2.2.25	2.5	3.7	Low

System/Evolution Name: Generic

Category Statement: Equipment Control

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

UserID: Topic

Question Stem:

Limits on RCS activity provided in Technical Specifications are based on the dose that would be received at the site boundary in a SGTR accident that begins with a steady-state primary-to-secondary leakage of 1 gpm.

Maintaining those limits ensures that the 2-hour dose at the site boundary during a SGTR will NOT exceed ________________.

A 10 CFR 20, Standards for Protection Against Radiation, limits.

B 10 CFR 100, Reactor Site Criteria, limits.

C EPA Protective Action Guideline thresholds.

D 5 Rem TEDE for the general public.

Answer:	Task No:		Question Source:	Question Difficulty
B	Obj No:	S.RC1-14, S.BZ1-03	Byron NRC exam bank (2000)	Medium

Time: 1

Cross Ref:

Reference:

I1-BZ-XL-01, Ch 2, Site and Buildings, Pgs 5, 6.
I1-RC-XL-01, Ch 12, Reactor Coolant System, Pg 51.
Tech Spec 3.4.16, RCS Activity, and 3.4.13, RCS Operational Leakage, bases
I1-SG-XL-01, Ch 22, Steam Generators, Pg 19

Explanation:

This is the bases statement for the RCS Activity limit in Tech Spec 3.4.16 and the RCS Operational Leakage Limit bases in Tech Spec 3.4.13. It is also taught as the document used to establish site boundary guidelines in Site and Building lesson plan.
A. is incorrect, but plausible since most controlled release limits are found in this document.
B. is correct as stated above.
C. is incorrect, but plausible since evacuation actions are based on this guidline.
D. is incorrect, but plausible since this is the federal limit for rad workers.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
70	Both	3		194001	2.2.34	2.8	3.2	High

System/Evolution Name: Generic

Category Statement: Equipment Control

KA Statement:
Knowledge of the process for determining the internal and external effects on core reactivity.

UserID: Topic

Question Stem:

A Unit 2 reactor startup is in progress following a refueling outage.

- Moderator Temperature Coefficient (MTC) is slightly positive at +.5 pcm/°F
- With critical data just obtained, rods are withdrawn to establish a +.1 DPM steady-state startup rate.

With no additional operator action, reactor power will RISE until which of the following occurs?

A The RCS heatup is sufficient to negate the reactivity added by the rods.

B Fuel temperature rises sufficiently to negate the reactivity added by the rods.

C Automatic steam dump response lowers RCS temperature.

D An automatic reactor trip occurs.

Answer: B

Task No:

Obj No: A.RT3-08/15

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RT-XL-03, Ch 3, Reactor Theory, MTC and Total Power Defect, Pgs 9, 19, 33.

Reference:

Explanation:

MTC is + which means the point of adding heat will then add additional + reactivity as the coolant temperature rises. However, FTC is always negative and fuel temperature rise must occur to raise RCS temperature. The magnitude of negative reactivity added by the fuel temperature rise will rapidly offset the positive reactivity added by the moderator temperature rise. FTC is ~ 3 times as large in magnitude as MTC.
A is incorrect since RCS heatup alone would continue to add + reactivity and power would continue to rise.
B. is correct as described above.
C. is incorrect since this would raise steam demand, raising power, holding temperature, not cooling down. Actual stabilization from FTC feedback to hold power.
D. is incorrect since power will turn and stabilize low into the power range. Critical data is taken well above P-6 and stablization will be well below P-10.
.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
71	Both	3		194001	2.3.2	2.5	2.9	Low

System/Evolution Name:
Generic

Category Statement:
Radiological Controls

KA Statement:
Knowledge of facility ALARA program.

UserID: Topic

Question Stem:

Given the following conditions on Unit 2:

- Operators are in the process of removing an OOS on a valve located in a high radiation area.
- Rad Protection estimates that performing an independent verification will cause an individual to receive 26 mrem.

What is the requirement for independent verification during this evolution?

A Must be performed unless waived by the Operations Manager.

B Operator self-check is substituted whenever an accumulated dose > 20 mrem is involved.

C Is NOT required with Shift Manager approval. However, alternate verification techniques shall be considered.

D The verifier should position themselves, allowing the best view but lowest dose, and observe the positioning and self-check of the first operator.

Answer: C

Task No: HU-002

Obj No:

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference: Conduct of Operations, BAP 300
HU-AA-101, section 4.3.1.1

Explanation:
The independent verification requirements are given in HU-AA-101 and this issue is specifically addressed and states that in this circumstance, the Shift Manager can make a determination for ALARA concerns to NOT perform a hands-on independent verification, but shall consider alternate methods of verification, such as flow, pressure, indicating lights, etc.
A. is incorrect since the procedure specifies the Shift Manager.
B. is incorrect since this is not left for the operator to decide.
C. is correct as described above.
D. is incorrect since this does NOT constitute an independent verification. This is close to concurrent verification.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
72	Both	3		194001	2.3.4	2.5	3.1	High

System/Evolution Name: Generic

Category Statement: Radiological Controls

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

UserID:

Topic

Question Stem:

A reactor operator has a Deep Dose Equivalent (DDE) of 1.3 Rem and a Committed Effective Dose Equivalent (CEDE) of 0.6 Rem to date.

With management approval, what is the maximum additional CEDE dose allowed during this year for this reactor operator? (Assume NO change in DDE dose for the remainder of the year)

A 1.1 Rem.

B 3.1 Rem

C 3.7 Rem

D 4.4 Rem

Answer: B

Task No:

Obj No: 6, 7, 15, 16

Question Source: Prairie Island NRC exam bank (1996)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
RP-AA-203, section 4.1.1
RWT, Exelon Radiation Worker Training, Pgs 5, 1, 12

Explanation:
DDE+CEDE=TEDE and the federal limit is 5 Rem/yr. 1.3 Rem + .6 Rem = 1.9 Rem. The Admin limit is 2 Rem, but with management approval dose to the federal limit can be authorized. Therefore, 1.9 Rem + 3.1 Rem = 5 Rem, so 3.1 Rem is correct.
A. is incorrect since this would = 3 Rem total, which used to be an old quarterly limit.
B. is correct as described above.
C. is incorrect, but plausible if it was thought DDE was the only dose that was considered part of TEDE.
D. is incorrect, but plausible if it was thought CEDE was the only dose that was considered part of TEDE.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
73	Both	3		194001	2.4.17	3.1	3.8	Low

System/Evolution Name:
Generic

Category Statement:
Emergency Procedures/Plan

KA Statement:
Knowledge of EOP terms and definitions.

UserID:

Topic

Question Stem:

A LOCA has occurred on Unit 2 with the following Containment conditions:

TIME	CNMT Pressure	CNMT Radiation
(1)-1000	3.4 psig	5.5 E4 R/hr
(2)-1005	4.8 psig	1.1 E5 R/hr
(3)-1010	8.2 psig	6.2 E5 R/hr
(4)-1015	4.9 psig	5.0 E5 R/hr
(5)-1020	2.0 psig	4.5 E4 R/hr

Based on the above parameters being addressed at the designated times during this event, When did Containment FIRST go ADVERSE and when can NORMAL values be used?

	FIRST GO ADVERSE	NORMAL values can be used
A	2	1
B	3	1, 5
C	3	4, 5
D	2	1, 4, 5

Answer: A

Task No:

Obj No: T.EP01-04

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Procedure use and Adherence

Reference:
I1-EP-XL-01, Reactor Trip or Safety Injection, Pg 7

Explanation:

Adverse CNMT conditions are defined as pressure above 5 psig or rad levels > 1E5 R/hr. The pressure component portion will not permanently impact the instrumentation and normal values may resume when pressure drops below 5 psig. The radiation component may cause permanent damage to the instrumentation and requires an engineering evaluation before normal values can be resumed regardless of how low the rad levels drop after reaching the adverse limit.

A. is correct since rad levels are first > 1E5 R/hr and time 1 is the only time that allowed use of normal values.
B. is incorrect since rad levels were above the limit at time 2 and 5 would not be allowed since rad levels had exceeded 1E5 R/hr.
C. is incorrect since rad levels were above the limit at time 2 and 4 or 5 do not allow normal use after rad levels were > 1E5 R/hr.
D. is incorrect since time 1 is the only time normal values could have been used.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref: None

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
74	Both	3		194001	2.4.22	3.0	4.0	Low

System/Evolution Name:
Generic

Category Statement:
Emergency Procedures/Plan

KA Statement:
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

UserID: Topic

Question Stem:

What are the priorities of the Byron Status Trees based on?

A Ensures symptoms are addressed so as many fission product barriers as possible are intact at all times to insure health and safety of the public.

B Ensures critical safety functions are addressed in the same sequence as the analyzed events in the UFSAR and Byron Emergency Procedures.

C Ensures the hierarchy is sequenced to protect plant personnel by addressing the RCS boundary, then the fuel boundary, then the Containment boundary.

D Ensures events are addressed in a sequence of MOST likely barrier to be lost to LEAST likely barrier to be lost to protect public health and safety.

Answer: A

Task No:

Obj No: T.FR7-03

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-FR-XL-07, Status Trees, Pgs 6, 7

Reference:

Explanation:

As stated in the background documents and the Lesson Plan, BSTs are symptom based, not event based and prioritized to ensure as many barriers as possible at all times to protect public health and safety. The hierarchy is sequenced to address fuel first, RCS second and containment third.
A. is correct as stated above.
B. is incorrect because the analysis sequence has nothing to do with the bases and the bases is not event based.
C. is incorrect because bases is to protect public first and the barrier sequence is not correct.
D. is incorrect since this is not how BSTs are sequenced and they are symptom based, not event based.

Date Written: 3/16/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
75	Both	3	4	94001	2.4.39	3.3	3.1	Low

System/Evolution Name:
Generic

Category Statement:
Emergency Procedures/Plan

KA Statement:
Knowledge of the RO's responsibilities in emergency plan implementation.

UserID: Topic

Question Stem:

What is the MAXIMUM time that may elapse, after emergency event classification, before State and County agencies must be notified?

A 15 minutes.

B 30 minutes.

C 1 hour.

D 4 hours.

Answer: A

Task No:

Obj No: G-5-8/9

Question Source: Kewaunee NRC exam bank (1996)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
Emergency Preparedness, G-5, Classification, Notification, and PARs, Pgs 17, 18
EP-AA-1002, Byron Annex for Radiological Emergency Plan, pg 3-10

Explanation:
Per the Byron Annex, and reportability manual, any classification exceeding EALs requires that State and Local authorities be notified within 15 minutes of the declaration to activate these outside agencies for event support to protect public health and safety.

A is correct as stated above.
B. is incorrect since this is not one of the notification times, but seems plausible.
C. is incorrect since this is the required time to notify the NRC of the event.
D. is incorrect since this would be a somewhat standard NRC notification time for many plant events that don't exceed EALs.

Date Written: 3/10/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
76	SRO	1	1	000008	2.1.32	3.4	3.8	High

System/Evolution Name:
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Category Statement:
Conduct of Operations

KA Statement:
Ability to explain and apply all system limits and precautions.

UserID:

Topic

Question Stem:

Unit 2 is at 80% power.

- PZR PORV 2RY456 lifted and stuck partially open causing an RCS pressure transient.
- Operators were unable to position the PORV manually and closed 2RY8000B, PORV Block valve.
- RCS pressure dropped to 2215 psig and is slowly trending to normal.

Which of the following is the NEXT required action and explains why the action is taken?

A Remove power from 2RY8000B within 1 hour to ensure positive control of the relief path while efforts to restore 2RY456 to OPERABLE status are in progress.

B Maintain 2RY8000B CLOSED and ENERGIZED to ensure the relief path is available with manual actions if needed.

C Place the control switch for 2RY456 in the CLOSED position within 1 hour to preclude automatic opening for an overpressure event at a time that the block valve is CLOSED.

D Place the unit in MODE 3 within 7 hours since Technical Specifications do not address the operability of a PZR PORV that is STUCK partially open.

Answer: A

Task No:

Obj No: S.RY1-26/28

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
Tech Spec 3.4.11 and Bases, PZR PORVs
I1-RY-XL-01, Ch 14, Pressurizer, Pgs 40, 41
43(b) (2)

Explanation:
Per Tech Spec 3.4.11, if a PORV is inoperable, the PORV Block valve must be closed and deenergized within 1 hour. This ensures positive control of the relief path per Tech Spec Bases and affords isolation for repair efforts to restore operability within 72 hours.

A. is correct as described above.
B. is incorrect since this action is done if the PORV is still capable of manual operation.
C. is incorrect since this is not the required action and it would have no affect on a stuck valve.
D. is incorrect since this is Tech Spec 3.0.3 action and does not apply. The inoperability is addressed by Tech Spec 3.4.11.

Date Written: 3/17/2006 Author: L. Wehner App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
77	SRO	1	1	000011	2.4.4	4.0	4.3	High

System/Evolution Name:
Large Break LOCA

Category Statement:
Emergency Procedures/Plan

KA Statement:
Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

UserID: Topic

Question Stem:

Given the following conditions on Unit 2:

- An SI has occurred.
- RCS pressure = 1200 psig and lowering.
- PZR level is off-scale low.
- CETC's = 535°F.
- Containment pressure = 9 psig and rising.
- Containment radiation monitors are in ALARM.
- All SG pressures = 775 psig and lowering slowly.

With these conditions, what event is occurring and what procedure would be entered after applicable steps of 2BEP-0, Reactor Trip or Safety Injection Unit 2, have been completed?

A A LOCA has occurred on an RCS loop cold leg, enter 2BEP-1, Loss of Reactor or Secondary Coolant Unit 2.

B ONE SG has a steamline break inside of containment, enter 2BEP-2, Faulted Steam Generator Isolation Unit 2

C A PZR PORV has fully opened and it's block valve is open, enter 2BEP-1, Loss of Reactor or Secondary Coolant Unit 2.

D A feedwater line break to ONE SG and MSIVs failed to close, enter 2BEP-2, Faulted Steam Generator Isolation Unit 2.

Answer: A

Task No:

Obj No: T.EP01-06-A,

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-EP-XL-01, 1/2BEP-0, Reactor Trip or Safety Injection, Pg 28
I1-EP-XL-02, Loss of Reactor or Secondary Coolant, Pg 7
43(b) (5)

Explanation:
The parameter that absolutely determines RCS vs steam/feed line is containment rad monitors. RCS cold leg vs PZR PORV will be seen in PZR level (low=loop break, high indicates PORV or Safety open). The determination in 2BEP-0 at steps 27 and 29 will direct entering 2 BEP-1 based on these symptoms.
A. is correct based on the diagnostic in step 27 and 29 of 2BEP-0, low RCS press, high cnmt press and radiation, and low PZR level.
B. is incorrect since high rad exists in the cnmt and MSIVs would be closed which would identify a faulted SG by seeing a lower pressure in one SG after isloation.
C. is incorrect since this would have been looked at in step 24 of 2BEP-0. It would be considered a small break LOCA and with low PZR press and level and a large RCS cooldown would not be indicative of a stuck open PORV.
D. is incorrect for similar reasons described in B. above.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
78	SRO	1	1	000015	2.1.32	3.4	3.8	High

System/Evolution Name:
Reactor Coolant Pump (RCP) Malfunctions

Category Statement:
Conduct of Operations

KA Statement:
Ability to explain and apply all system limits and precautions.

UserID: Topic

Question Stem:

All RCP's were manually tripped at 1425 psig during an emergency procedure implementation. During the recovery, the TSC has directed the shift to start one RCP per BOP RC-1, STARTUP OF A REACTOR COOLANT PUMP.

Given the following information: (Assume other plant conditions are satisfactory to support RCP start)

	A RCP	B RCP	C RCP	D RCP
- Seal injection flow	8 gpm	10 gpm	9 gpm	10 gpm
- Seal leakoff flow	1.2 gpm	.9 gpm	.4 gpm	1.7 gpm
- #1 Seal D/P	330 psid	325 psid	198 psid	265 psid
- Motor Upper Radial Bearing Temp	144°F	145°F	132°F	147°F
- Motor Lower Radial Bearing Temp	198°F	153°F	149°F	144°F
- Motor Upper Thrust Bearing Temp	147°F	142°F	151°F	146°F
- Motor Lower Thrust Bearing Temp	112°F	108°F	107°F	106°F

Which RCP should the Unit Supervisor direct the NSO to start?
(BOP RC-1A1, RCP NO 1 SEAL LEAKOFF NORMAL OPERATING RANGE, is provided)

A RCP A

B RCP B

C RCP C

D RCP D

Answer: B

Task No:

Obj No: S.RC2-09-A/B

Question Source: Byron LORT exam bank - Used last 2001 cycle 6-6a quiz.

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
BOP RC-1, Startup of a Reactor Coolant Pump
BOP RC-1A1, RCP NO 1 Seal Leakoff Normal Operating Range
I1-RC-XL-02, Ch 13, Reactor Coolant Pump, Pgs 30-32
43(b)(5)

Explanation:
D RCP seal leakoff is > 1.4 gpm, and C RCP #1 Seal D/P is < 200 psid, leaves B or A RCPs. A RCP exceeds the 195°F limit for a motor bearing. The order of preferrence is D, C, B, A.

Date Written: 3/27/2006 Author: J. Heaton App. Ref: BOP RC-1A1

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
79	SRO	1	1	000025	AA2.07	3.4	3.7	High

System/Evolution Name:
Loss of Residual Heat Removal System (RHRS)

Category Statement:
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System:

KA Statement:
Pump cavitation

UserID: Topic

Question Stem:

Unit 2 has just completed refueling.

- Preparations are in progress to install the reactor vessel head.
- 2B RH pump is providing shutdown cooling with ~ 3300 gpm flow rate.
- 2A RH pump is running transferring refueling cavity water to the RWST per BOP RH-9, Pumpdown of the Refueling Cavity to the RWST.
- 2A RH pump has just been reduced to ~490 gpm, by throttling 2RH618, RH HX 2A Bypass Flow Control Vlv, with cavity level at ~ 403' elevation.

Flow and motor amps ocillations are reported on the 2B RH train.

What is the NEXT required action the Unit Supervisor will direct?

A Stop draining the refueling cavity by closing 2RH618 and TRIP the 2B RH pump, then enter 2BOA PRI-10, Loss of RH Cooling Unit 2.

B Stop draining the refueling cavity by closing 2RH618 and reduce flow through the 2B RH train in an attempt to stablize flow and amps per BOP RH-9.

C Trip BOTH 2A and 2B RH pumps and enter 2BOA S/D-2, Shutdown LOCA.

D Stop draining the refueling cavity by closing 2RH618 and trip the 2B RH pump, then place the 2A RH train in the shutdown cooling mode using BOP RH-6, Operations of RH System in Shutdown Cooling.

Answer: B

Task No:

Obj No: S.RH1-09-C

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-RH-XL-01, Residual Heat Removal System, Pgs 29-32
BOP RH-9, Pumpdown of the Refueling Cavity
43(b) (5)

Reference:

Explanation:

A CAUTION in BOP RH-9, prior to step F.1, states that if the above condition exist, stop draining immediately, reduce RH flow in the shutdown cooling train to stablize and initiate BOP RH-8 as necessary to establish adequate level for suction on the RH train in SDC. If the RH train in SDC will not stablize with the reduced flow and RH pump trip is required or the RH pump trips, go to 2BOA PRI-10.
A. is incorrect since tripping the 2B RH pump is not directed first. Attempt to stablize flow by reducing flow first.
B. is correct as stated in BOP RH-9.
C. is incorrect since tripping both pumps is not the immediate action and if you did, 2BOA S/D-2 is not correct.
D. is incorrect since this is action that may be directed in 2BOA PRI-10, but not in the current procedures.

Date Written: 3/17/2006 Author: L. Wehner App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
80	SRO	1	1	000027	AA2.04	3.7	4.3	Low

System/Evolution Name:
Pressurizer Pressure Control (PZR PCS) Malfunction

Category Statement:
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:

KA Statement:
Tech-Spec limits for RCS pressure

UserID: Topic

Question Stem:

Following a refueling outage, Unit 2 is at 40% power with a power ascension at maximum preconditioning rate in progress. The Master PZR Pressure Controller, 2PK-455A, was discovered failing high. When the operator placed the controller in MANUAL, the following conditions were present:

- RCS Tave = 567°F.
- PZR pressure = 2175 psig.
- PZR level = 33%.
- CVCS letdown is isolated.
- Excess letdown is in service.

How will the RCS DNB limits be addressed under these conditions?

A PZR pressure must be raised to at least 2209 psig within 2 hours.

B PZR level must be restored to within 5% of program within the next 2 hours.

C NO action is required since PZR pressure limit is NOT challenged at this power level.

D NO action is required since RCS temperature limit is NOT exceeded.

Answer: A

Task No:

Obj No: S.RC1-12

Question Source: Byron NRC exam bank (1998)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
Tech Spec 3.4.1. RCS Pressure, Temperature, and Flow DNB Limits. And COLR
I1-RC-XL-01.Reactor Coolant System, Pg 41
NF-AP-440, Fuel Preconditioning Limits
43(b)(2)

Explanation:
Above 40% power, fuel preconditioning limits apply and the ramp rate is limited to much < 5%/minute, therefore, the PZR pressure below 2209 psig Tech Spec action to restore within 2 hours applies. DNB limit for Tave is 593.1°F, therefore 567°F is well below that limit. PZR level at 33% is well below program level of ~ 41%, but there is no 2 hour requirement to restore within 5% of program. This would be done in response to a PZR level deviation alarm that would be in. Which letdown system is in service has no bearing on the isssue here.
A. is correct as explained.
B. is incorrect as explained.
C. is incorrect since this is applicable when the ramp is < 5%/minute.
D. is incorrect for the PZR pressure, even though true if only based on RCS temperature.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
81	SRO	1	1	00WE12	EA2.1	3.2	4.0	High

System/Evolution Name:
Uncontrolled Depressurization of all Steam Generators

Category Statement:
Ability to determine and interpret the following as they apply to the Uncontrolled Depressurization of all Steam Generators:

KA Statement:
Facility conditions and selection of appropriate procedures during abnormal and emergency operations

UserID: Topic

Question Stem:

Unit 1 was at 100% power when the following events occurred:

- ALL SGs are faulted into containment.
- While performing steps in 1BCA-2.1, Uncontrolled Depressurization of All Steam Generators, a RED path is noted on the containment critical safety function.
- Actions of 1BFR-Z.1, Response to High Containment Pressure, are performed.
- Auxiliary Feedwater has been throttled to 45 gpm to each SG.
- When directed by 1BFR-Z.1 to return to procedure and step in effect, the following is noted on the CSF status trees:

Subcriticality	- Green
Core Cooling	- Green
Heat Sink	- Red
Integrity	- Orange
Containment	- Red
Inventory	- Yellow

It is required to ENTER AND PERFORM STEPS in which of the following procedures NEXT?

A 1BEP-2, Faulted Steam Generator Isolation.

B 1BFR-H.1, Response to Loss of Secondary Heat Sink.

C 1BFR-P.1, Response to Imminent Pressurized Thermal Shock.

D 1BFR-Z.1, Response to High Containment Pressure.

Answer: C

Task No:

Obj No: T.FR7-03/07,

Question Source: Byron NRC exam bank (2000) - modified

Question Difficulty: Medium

Time: 1

Cross Ref:
I1-FR-XL-07, Status Trees, Pg 9,
I1-FR-XL-03, BFR H.1-H.5, Pg 5
43(b)(5)

Reference:

Explanation:
Once 45 gpm has been established to the faulted SGs in 1BCA-2.1, the CAUTION (If total feed flow is < 500 gpm due to operator action, this procedure should NOT be performed) prior to the first step in 1BFR-H.1 will apply and the SRO will proceed to the next CSF in the hierachy, INTEGRITY, since CONTAINMENT was just addressed and INTEGRITY is orange.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
82	SRO	1	2	000069	AA2.01	3.7	4.3	High

System/Evolution Name:
Loss of Containment Integrity

Category Statement:
Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:

KA Statement:
Loss of containment integrity

UserID: Topic

Question Stem:

A small break LOCA has occurred on Unit 2.

- Manual reactor trip and SI was actuated.
- All systems responded as expected.
- Estimated leakrate is ~ 400 gpm.
- RCS pressure is 2000 psig and slowly rising.
- PZR level is 22% and rising.
- Containment radiation is 1.5 R/hour and stable.
- Containment pressure reached 9.0 psig, then rapidly dropped to 0.5 psig.

What is the current classification for this event based on these indications?
(Byron Annex Attached)

A Unusual Event - FU1

B Unusual Event - MU8

C Alert - FA1

D Site Area Emergency - FS1

Answer: D

Task No:

Obj No: T.ZP1-16A, G-6-8

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

EP-AA-1002, Byron Annex for Radiological Emergency Plan
Emergency Preparedness, G-6, Emergency Action Levels
43(b)(5)

Reference:

Explanation:

Applying the indications above will lead you to addressing the fission product barriers. RCS should be identified as potentially lost, due to leakrate > 1 CV pump in normal operation, and that alone would be classified as an ALERT-FA1. The containment should be identified as lost due to the rapid, unexplainable, pressure drop. This alone would be an Unusual Event - FU1. If leakage is thought to be within the capacity of 1 CV pump in normal lineup, then an Unusual Event - MU8 may be declared. The combination of the RCS leakage and the containment rapid depressurization together would be a Site Area Emergency - FS1. This would be correct for the event conditions.
A. is incorrect since more than just the containment barrier is lost or potentially lost.
B. is incorrect since the leakrate applies, but it is large enough to constitute potential failure of the RCS.
C. is incorrect since more than the RCS barrier is lost or potentially lost.
D. is correct as described above.

Date Written: 3/19/2006 Author: M. Jorgensen App. Ref: Byron Annex for Emergency Rad Plan

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
83	SRO	1	2	000076	AA2.01	2.7	3.2	High

System/Evolution Name:
High Reactor Coolant Activity

Category Statement:
Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity:

KA Statement:
Location or process point that is causing an alarm

UserID: Topic

Question Stem:

At 0800 - Unit 2 was performing a power ascension from 75% to 90% after a Refueling Outage.
- Excessive Iodine Spiking is reported by Chemistry.

Which Radiation Monitor would first detect the Iodine Spiking and what action would be taken?

A VCT Cubicle Monitor; direct RP to survey the Auxiliary Building and establish required area postings.

B CNMT ATMOS Monitor; verify CNMT Ventilation Isolation and consider placing a CNMT Charcoal filter train in service.

C MAIN STEAMLINE Monitor; verify a single Charging pump is adequate to maintain PZR level > 17%.

D GROSS FAIL FUEL Monitor; contact Chemistry to calculate mixed bed demin decontamination factor.

Answer: D

Task No:

Obj No: T.OA15-03/05

Question Source:

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-OA-XL-15, BOA PRI-4, Abnormal Primary Chemistry, Pg 5

Reference:
1/2BOA PRI-4, Abnormal Primary Chemistry, step 7
43(b) (5)

Explanation:
It is not unusual for some Iodine Spiking to occur on a rapid power change and the activity will usually return to normal levels after a few hours of operation. One BOA PRI-4 entry condition is High alarm on the Gross Failed Fuel monitor.The first action to be taken is to have Chemistry calculate the Demin DF to ensure the activity can be removed..
A. is incorrect since this is a High range monitor that may trend up, but is only expected to alrm with very high activity in the VCT and very low level in the VCT.
B. is incorrect since there is nothing in the stem to lead one to believe a leak to atmosphere in Cnmt exists.
C. is incorrect since there is nothing in the stem to lead one to believe a SG tube leak exists.
D. is correct as described above.

Date Written: 3/20/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
84	SRO	1	2	00WE02	2.2.22	3.4	4.1	High

System/Evolution Name:
SI Termination

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID: Topic

Question Stem:

A small break LOCA has occurred on Unit 1 and transition to 1BEP-1, Loss of Reactor or Secondary Coolant, has been made. The crew is at step 6 which determines if ECCS flow can be reduced. The following conditions exist:

- Containment pressure = 6.4 psig
- PZR level = 14%
- NR SG levels: 1A = 28%; 1B = 29%; 1C = 33%; 1D = 32%
- AF flow to SGs = 590 gpm
- RCS pressure = 1482 psig and stable
- CETCs = 523°F
- All RCPs are running
- RWST level = 49%

Based on these conditions, ECCS __________________________.

A can be terminated, transition to 1BEP ES-1.1, SI Termination.

B can not be terminated due to inadequate PZR level, continue in 1BEP-1.

C can not be reduced due to RWST level, transition to 1BEP ES-1.3, Transfer to Cold Leg Recirculation.

D can not be reduced due to RCS subcooling, transition to 1BEP ES-1.2, Post LOCA Cooldown and Depressurization.

Answer: B

Task No:

Obj No: T.EP02-01-F

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
1BEP-1, Loss of Reactor or Secondary Coolant, Step 6 criteria
I1-EP-XL-02, Loss of Reactor or Secondary Coolant, Pg 8
1BEP-0, Reactor Trip or Safety Injection, CAUTION prior to step 1
43b(5)

Explanation:
Usage requirements and first CAUTION in 1/2BEP-0 for ADVERSE conditions (> 5 psig in containment) requires use in the conditions stated. PZR level must be > 28% for SI termination. All other conditions are met in step 6 of 1BEP-1.
A. is incorrect since PZR level (Adverse CNMT) is not adequate.
B. is correct as stated above.
C. is incorrect since RWST level is not a termination criteria and is above the LO-2 setpoint for transition to 1BEP ES-1.3.
D. is incorrect since subcooling is addequate.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
85	SRO	1	2	00WE09	2.4.30	2.2	3.6	High

System/Evolution Name:
Natural Circulation Operations

Category Statement:
Emergency Procedures/Plan

KA Statement:
Knowledge of which events related to system operations/status should be reported to outside agencies.

UserID: Topic

Question Stem:

Which of the following events would require notification of the NRC within 1 hour?

A Two vehicles crash in the Byron Station parking lot with minor injuries; no one hospitalized.

B Off-site power is lost to Unit 2 at 18% power resulting in a reactor trip.

C SAT feeder breaker to Bus 141 trips and 1A DG fails to start with Unit 1 at 100% power.

D A tornado is sighted within 5 miles of Byron Station.

Answer: B

Task No:

Obj No: G-6_8, G-5-8/9

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
Emergency Preparedness, G-6, Emergency Action Levels, Pg 15
Emergency Preparedness, G-5, Classification, Notification, and PARs, Pgs 17, 18
EP-AA-1002, Byron Annex For Radiological Emergency plan, Pg 3-10
43(b) (5)

Explanation:
Per the Byron Annex, the loss of a single ESF Bus nor the reactor trip alone exceed EALs, (SM could declare an NUE, but not required). This is reportable to the NRC, but not within 1 hour. However, the loss of both SATs places Unit 2 in a natural circ condition and means the SATs are not available to power the Unit 2 ESF buses. This meets EAL MU1 and any classification requires 15 minute notification of State and Local authorities and notification of the NRC within 1 hour.
A. is incorrect since this event does not exceed an EAL, crash inside of the protected area or swithyard into structures containing systems for sfe shutdown would exceed the EAL and require notification.
B. is correct as described above.
C. is incorrect since this would be reportable, but not within 1 hour and does not exceed an EAL.
D. is incorrect since a tornado strike within the protected area or switchyard is required to exceed an EAL.

Date Written: 3/22/2006 Author: M. Jorgensen App. Ref: PE-AA-1002, Byron Annex

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
86	SRO	2	1	012000	A2.06	4.4	4.7	Low

System/Evolution Name:
Reactor Protection System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Failure of RPS signal to trip the reactor

UserID: Topic

Question Stem:

Unit 2 operators are performing steps in 2BFR S.1, Response to Nuclear Power Generation/ATWS Unit 2, and have just isolated steam dumps when the following conditions are noted:

- Unit 2 reactor power = 19%.
- PZR pressure is 1830 psig and trending down.
- Reactor trip breakers are closed.
- Safety injection just Actuated.

What actions will you direct for these conditions?

A Continue attempts to trip the reactor and transition to 2BEP-0, Reactor Trip or Safety Injection, for SI Verification.

B Verify SI is actuated per the Operator Action Summary page in 2BFR S.1 while continuing with the steps in 2BFR S.1.

C Continue in 2BFR S.1 and after the reactor is tripped, immediately transition to 2BEP-0.

D Verify SI is properly actuated using the verification steps in 2BEP-0 and then continue with the steps of 2BFR S.1.

Answer:	Task No:		Question Source:	Question Difficulty
B	Obj No:	T.FR1-07	Byron LORT Bank. (2002) - Used last in 2004, cycle 6-8 SRO annual.	Medium

Time: Cross Ref:
1

1BFR S.1, Response to Nuclear Power Generation/ATWS Unit 2 Reference:
I1-FR-XL-01, BFR S Series Subcriticality, Pg 21
43(b)(5)

Explanation:
Since 2BFR S.1 could have been entered at step 1 of 2BEP-0 and SI may have occurred and not been verified, the contingency is covered separately in 2BFR S.1 after step 5 in a CAUTION stating "If SI actuates, proper ESF actuations should be verified as time permits per the OAS page.
A. is incorrect since transition out of this procedure is not done after step 6 until the procedure is completed. Step 7 has just been completed by the crew.
B is correct as stated above.
C. is incorrect for essentially the same reason as A. above.
D. is incorrect since the OAS page of 2BFR S.1 contains the verification steps and 2BEP-0 is not required.

Date Written: 3/22/2006 Author: G. Wolfe App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
87	SRO	2	1	026000	2.2.22	3.4	4.1	Low

System/Evolution Name:
Containment Spray System (CSS)

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID: Topic

Question Stem:

What is the safety analysis basis for the minimum OPERABILITY requirements for the Spray Additive System?

The Design Basis Accident analyses assumes that ____________________.

A ONE train is OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the Lo-2 setpoint is adequate to ensure a minimum 8.0 pH in the containment recirculation sump to reduce stress corrosion of mechanical components.

B ONE train is OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches Lo-3 setpoint is adequate to remove iodine from the containment atmosphere and maintain it in solution in the recirculation sump.

C TWO trains are OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the Lo-3 setpoint is adequate to ensure a minimum 8.0 pH in the containment recirculation sump to reduce stress corrosion of mechanical components.

D TWO trains are OPERABLE and the volume of NaOH added to the spray in the time the RWST reaches the Lo-2 setpoint is adequate to remove iodine from the containment atmosphere and maintain it in solution in the recirculation sump.

Answer:	Task No:		Question Source:	Question Difficulty
B	Obj No:	S.CS1.03/15	Byron Cert exam bank (2001)	Medium

Time: 1

Cross Ref:

Tech Spec 3.6.7 bases, Pgs B3.6.7-2/3

Reference:
I1-CS-XL-01, Ch 59, Containment Spray System, Pgs 2, 3, 23, 24
43(b) (3)

Explanation:

As stated in the Tech Spec bases; only ONE train required until CS is switched to the recirculation sump when the RWST is empty (Lo-3) and removes iodine and maintains it in solution with the pH adjustment. This, in turn, minimizes corrosion of components.
A. is incorrect since the DBA assumes CS runs at least to the Lo-3 setpoint and the premise for pH is iodine removal.
B. is correct as stated above.
C. is incorrect since the design assumes only a single train operates and the pH reasoning is not complete.
D. is incorrect since the design assumes only one train operates to Lo-3 in the RWST.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
88	SRO	2	1	059000	2.2.22	3.4	4.1	Low

System/Evolution Name:
Main Feedwater (MFW) System

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID:

Topic

Question Stem:

The following conditions exist on Unit 2:

- MODE 3 with reactor startup planned.

An NLO reports the following FW valves packing was adjusted due to excessive leakage:

- 2FW002C, 2C FW PP MOV DISCH
- 2FW035A, S/G 2A FW TEMPERING ISOL VALVE
- 2FW043D, S/G 2D FWIV BYPASS ISOL VALVE

Which valves will you assign to have valve stroke time tests completed for Tech Spec 3.6.3, Containment Isolation Valves Operability?

A 2FW002C and 2FW043D only.

B All three valves.

C 2FW002C and 2FW035A only.

D 2FW035A and 2FW043D only.

Answer: D

Task No:

Obj No: S,CD1-021/022

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
Tech Spec 3.6.3 Bases, Containment Isolation Valves, Table B. 3.6.3-1 (page 3 of 9)
I1-CD-XL-01, Condensate and Feedwater System, Pg 63

43(b) (2)

Explanation:
Tech Spec 3.6.3, Containment Isolation Valves applies to 2FW035A and 2FW043D only. 2FW002C will also require testing since auto closure signals are sent to the valve, but it is not a containment isolation valve.
A. is incorrect since 2FW002C is not required in Tech Spec 3.6.3, it is not a cnmt isol valve.
B. is incorrect for the same reason as A.
C. is incorrect for the same reason as A.
D. is correct as described above.

Date Written: 3/22/2006 Author: G. Wolfe App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
89	SRO	2	1	062000	A2.03	2.9	3.4	High

System/Evolution Name:
A.C. Electrical Distribution System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the A.C. Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Consequences of improper sequencing when transferring to or from an inverter

UserID: Topic

Question Stem:

Unit 2 was at 100% power when the following occurred:

- The Inverter for Instrument Bus 211 failed resulting in loss of Bus 211.
- Operators responded per 2BOA ELEC-2, Loss of Instrument Bus Unit 2 and reenergized Bus 211 via the Constant Voltage Transformer (CVT).
- PR channel N44 is currently in a tripped condition due to detector failure requiring a shut down to repair
- 2 days later, maintenance has repaired the inverter and is ready to place it back in service.

What actions need to be directed BEFORE restoring Instrument Bus 211 to the Inverter?

A Leave PR channel N44 as is and restore power to the Instrument bus from the Inverter to the CVT per 2BOA ELEC-2.

B Place PR channel N44 in Bypass first, then restore power to the Instrument bus from the Inverter per BOP IP-1, Instrument Bus Inverter Startup.

C Place PR channel N41 on its alternate power supply, then restore power to the Instrument bus from the Inverter per BOP IP-1, Instrument Bus Inverter Startup.

D Place PR channel N41 in Bypass first, then restore power to the Instrument bus from the Inverter per 2BOA ELEC-2.

Answer: B

Task No:

Obj No: S.AP1-14-B

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

I1-AP-XL-01, Ch 4, AC Electrical Power Systems, Pgs 52, 53, 63, 64
BOP IP-1, Instrument Inverter Startup, Precautions
43(b)(5)

Reference:

Explanation:

BOP IP-1 has a precaution that reminds the operator that when transferring power from the reserve power to the inverter, the circuit is a break before make, resulting in a momentary loss of power, which may cause a reactor trip. So, if another instrument bus channels' instrumentation has a standing trip condition, this momentary loss of this channel may satisfy reactor trip logic.
A. is incorrect since this may result in a trip during transfer when power is momentarily lost to N41.
B. is correct since this is a trip condition on NI channel 4(N44) and if power is momentarilly lost to NI channel 1(N41) during the transfer back to the Inverter, this could result in a reactor trip(2/4 logic), if N44 is not taken to Bypass first.
C. is incorrect since N44 does not have a backup power supply like many of the SSPS instruments do.
D. is incorrect since this action alone could result in a reactor trip with N44 already in trip.

Date Written: 3/22/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
90	SRO	2	1	073000	A2.01	2.5	2.9	High

System/Evolution Name:
Process Radiation Monitoring (PRM) System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Erratic or failed power supply

UserID: Topic

Question Stem:

Unit 1 is at 100% power with the following conditions:

- 0100 on 6/5 - 1FS-RF008, Containment Floor Drain Sump Flow monitor, failed.
  - Parts have been ordered for replacement.

- 1300 on 6/30 - 1PR011J, CNMT ATMOS Unit 1, monitor has just alarmed indicating an OPERATE FAILURE.

What actions are/were directed as a result of these conditions?

A Enter LCOAR for 1FS-RF008 failure; enter LCO 3.0.3 for 1PR011J failure.

B Place 1FS-RF008 in the Degraded Equipment Log; enter LCOAR for 1PR011J failure.

C Enter LCOAR for 1FS-RF008 failure; enter separate LCOAR for 1PR011J failure.

D Place 1FS-RF008 and 1PR011J in the Degraded Equipment Log; verify alternate instumentation is OPERABLE.

Answer: B

Task No:

Obj No: S.RC1-12, S.AR1-06/15

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-RC-XL-01, Ch 12, Reactor Coolant System, Pg 50
I1-AR-XL-01, Ch 49, Radiation Monitoring, Pg 35
Tech Spec 3.4.15, RCS Leakage Detection Instrumentation
BAP 1400-6, Tech Spec LCOARs
43(b) (5)

Explanation:
Tech Spec 3.4.15 requires FS-RF008 or PC002 or PC003 AND FS-RF010, AND 1PR011J (A-Particulate channel) to be OPERABLE for RCS Leak Detection. With FS-RF008 OOS, the LCO is still met and, since FS-RF008 is part of the operability requirement, BAP 1400-6 requires that it be entered in the Degraded Equipment Log for tracking with no LCOAR required unless additional sump flow or level instrumentation fails. However, 1PR011J is required to be OPERABLE to meet the LCO, therefore, a LCOAR is required for its failure. This is a 30 day action requirement with additional sampling and surveillance required due to its failure. LCO 3.0.3 would be entered if 1PR011J and less than required sump flow/level indications were available.
A. is incorrect since no LCOAR is required for 1FS-RF008 alone failed, therefore LCO 3.0.3 would not apply.
B. is correct as described above.
C. is incorrect since no LCOAR is required for 1FS-RF008 alone failed and this Tech Spec does not allow separate entries for clock starts on LCOAR actions.
D. is incorrect since only 1FS-RF008 is required in the DEL and LCOAR is required for 1PR011J failure.

Date Written: 3/27/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
91	SRO	2	2	034000	K1.04	2.6	3.5	Low

System/Evolution Name:
Fuel Handling Equipment System (FHES)

Category Statement:
Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems:

KA Statement:
NIS

UserID: Topic

Question Stem:

The Fuel Handling Supervisor has just notified the Control Room that audible counts in CNMT ceased as they were lifting a fuel assembly from the upender. The NSO also reports that audible counts in the Control Room have been lost. The fuel handlers have suspended core alterations.

What is the requirement prior to resuming core alterations?

A Set the audible count rate selector switch to the other channel, VERIFY audible count rate to the Control Room and CNMT with ONE SR channel OPERABLE.

B Initiate emergency boration until it can be VERIFIED that all filled portions of the RCS are at least 2300 ppm boron concentration.

C Set the audible count rate selector switch to the other channel, VERIFY audible count rate to the Control Room and CNMT with BOTH SR channels OPERABLE.

D Set the audible count rate selector switch to the other channel, VERIFY audible count rate to the Control Room with ONE SR channel OPERABLE.

Answer: C

Task No:

Obj No: T.OA10-03

Question Source: Byron NRC exam bank (1996)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
1/2BOA INST-1, Nuclear Instrumentation Malfunction, step 3 and 6 for SR channels
Tech Spec 3.9.3 requires at least 2 operable for core alterations.
43(b) (7)

Explanation:
At least 2 operable channels are required for core alterations per Tech Spec 3.9.3. Also _BOA INST-1 action is to select the other channel and ensure that at least 2 channels are operable to perform core alterations or any + reactivity additions.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
92	SRO	2	2	045000	2.2.22	3.4	4.1	High

System/Evolution Name:
Main Turbine Generator (MT/G) System

Category Statement:
Equipment Control

KA Statement:
Knowledge of limiting conditions for operations and safety limits.

UserID: Topic

Question Stem:

Unit 2 is at 100% power with the following condition:

- NLO reports from the field that 2ES017A, Extraction Steam Nonreturn Check Valve, from the 24A Feedwater Heater, did not move during performance of the monthly surveillance.

What are the required actions for continued operations and why?
(BOP HD-6T1, Turbine Operations Limitation Table Concerning the Isolation of Various Strings of Feedwater Heaters, is provided)

A Reduce turbine load to 820 Mwe, then open the Low Pressure Heater String Bypass Valve and isolate the 21A, 22A, 23A, and 24A Feedwater Heater string to prevent possible turbine overspeed on a turbine trip.

B Reduce turbine load to 1173 Mwe, then isolate extraction steam to the 24A Feedwater Heater

to prevent possible turbine overspeed on a turbine trip.

C Reduce turbine load to 1173 Mwe, then isolate the MOVs in the extraction steam lines to the 24A/B/C Feedwater Heaters to preclude water induction into the turbine on a High level in the associated heaters.

D Reduce turbine load to 937, then isolate the MOVs in the extraction steam lines to the 22A, 23A, and 24A Feedwater Heaters to prevent possible turbine overspeed on a turbine trip.

Answer: B

Task No:

Obj No: S.ES1-04/11

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-ES-XL-01, Ch 36, Extraction Steam, Feedwater Heater Vents and Drains, Pgs 19, 20, 24, 32, 35
I1-MT-XL-01, Main Turbine and Reheat Steam
TRM 3.3.g
BOP HD-6T1, Turbine Operations Limitation Table Concerning the Isolation of Various Strings of Feedwater Heaters
BOP HD-16, Isolating and Return to Service LP FW Heaters _4A/B/C Shell Side and Restoration Following HI-2 Level Isolation
43(b) (2)

Explanation:
TRM 3.3.g, Condition D applies. Referring to Table 3.3.g-2, the action is to isolate the steam supply to the turbine within 6 hours. In the case of extraction steam, this is referring to the MOV (for the valve in question since no manual valve exists for this valve) in the extraction steam line, this isolates return steam to the turbine on a turbine trip, which then prevents the possibility of overspeed. Per BOP HD-6T1, power reduction to 1173 Mwe is required to isolate this nonreturn valve. Only this line is required to be isolated and operations can continue per BOP HD-16. The string or any other heater is not required to be isolated with it.
A. is incorrect since this is the required power reduction for removing the entire string from service, but this is not required.
B. is correct as described above.
C. is incorrect since all three heaters are not required to be isolated.
D is incorrect since the power reduction required is 1173. This would be required to remove the 3 heaters in the string, but that is not required.

Date Written: 3/24/2006 Author: M. Jorgensen App. Ref: BOP HD-6T1

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
93	SRO	2	2	075000	A2.01	3.0	3.2	High

System/Evolution Name:
Circulating Water System

Category Statement:
Ability to (a) predict the impacts of the following malfunctions or operations on the Circulating Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:

KA Statement:
Loss of intake structure

UserID: Topic

Question Stem:

With both Units at 100% power, the control room operators have identified a loss of flow condition on CW Makeup flow indicator, 0FT-CW040, in response to INTAKE BAY LEVEL LOW alarm.

If 0AOV-CW220, CW Makeup Flow Control Valve, does NOT open, the operators are procedurally directed to conserve flume level.

Which of the following actions would the operators be directed to perform?

- 1. Reduce reactor power and shutdown a CW pump on each unit.
- 2. Secure CW blowdown.
- 3. Secure CW makeup to SX.
- 4. Reduce reactor power on each unit.

A 2 and 4

B 2 and 3

C 1 and 3

D 1 and 2

Answer: A

Task No:

Obj No: T.OA43B-03

Question Source: Byron LORT bank - Used last 2006, cycle 1-4.

Question Difficulty: Medium

Time: 1

Cross Ref:

0BOA SEC-11, Inadequate Ciculating Water Makeup Reference:
I1-OA-XL-43, Inadequate Circulating Water Makeup, Pgs 3, 4
43(b)(5)

OBOA SEC-11 actions for makeup failure is to stop CW blowdown and reduce power on both units to conserve flume level.

Explanation:
A. is correct as stated above.
B. is incorrect since SX makeup is not a viable option, since SX is required for safe shutdown.
C. is incorrect since shutdown of CW pumps would not conserve flume level, SX actually takes priority.
D. is incorrect since shutdown of CW pumps would not conserve flume level.

Date Written: 3/23/2006 Author: G. Wolfe App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
94	SRO	3		194001	2.1.7	3.7	4.4	High

System/Evolution Name: Generic

Category Statement: Conduct of Operations

KA Statement:
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

UserID:

Topic

Question Stem:

Given the following conditions on Unit 2:

- Reactor power - 31% and rising.
- RCS Tave = 556°F and slowly lowering.
- PZR pressure = 2175 psig and slowly lowering.
- PZR level = 29% and slowly lowering.
- Turbine load is stable.
- SG pressures = 940 psig and lowering.
- SG PORVs indicate closed.
- Steam Dumps indicate closed.
- Unit 1 Turbine Bldg Rounds calls the Control Room and reports steam on the 451' level in the Turbine building.

What action should be directed for these conditions?

Immediately trip the ______________.

A turbine and close the MSIVs.

B reactor and close the MSIVs.

C turbine and initiate safety injection.

D reactor and initiate safety injection.

Answer: B

Task No:

Obj No:

Question Source: Byron Cert exam bank (2001)

Question Difficulty: Medium

Time: 1

Cross Ref:

OP-AA-101-111-1001, Operations Philosophy Handbook
Conditions of an Operating License, 10CFR50
43(b)(5)

Reference:

Explanation:

The call here is really a condition of license. It is expected that an SRO will have public health and safety foremost in his mind, then comes personnel safety on-site. This is clearly an example were the plant is degrading and needs to be placed in a safe condition and then take action to protect plant personnel. The indication of a steam leak in the turbine building is the key to why the plant conditions are trending the way they are. Also, protecting plant personnel requires isolation of the steam source.
The proper action is to trip the reactor to place it in a safe condition. This will also generate a turbine trip in all cases. Then close the MSIVs to protect plant personnel and stop the plant transient at the same time. A turbine trip first is inappropriate, since this is then relying on the turbine tripping to generate a reactor trip, which should occur at this power level, but should not be relied on and closing the MSIVs will also stop steam to the turbine.
A. is incorrect since the reactor trip will generate a turbine trip and ensuring the reactor is shutdown takes priority.
B. is correct as described above.
C. is incorrect since this may trip the reactor, it will not islate the steamline break.
D. is incorrect since the reactor trip first is OK, but the SI will not stop the steam leak to protect personnel.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
95	SRO	3		194001	2.1.32	3.4	3.8	Low

System/Evolution Name: Generic

Category Statement: Conduct of Operations

KA Statement:
Ability to explain and apply all system limits and precautions.

UserID:

Topic

Question Stem:

LCO 3.0.6 requires a Loss of Safety Function Evaluation (LOSF) be performed when an inoperabilty of a support system renders a supported system inoperable.

During a situation involving two supported systems inoperable due to the same support system without a LOSF, to ensure further LOSF evaluations on the supported systems to be properly performed, per BAP 1400-6, "TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATION ACTION REQUIREMENTS (LCOAR)", the inoperable supported systems are REQUIRED to be logged in the ___________.

A Degraded Equipment Log (DEL)

B Shift Managers Log

C supported systems LCOAR paperwork and the associated Unit Log

D associated supported systems LCOAR

Answer: A

Task No:

Obj No: S.TS1-05-B, T.AM13-

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
I1-TS-XL-01, Ch 3, Introduction to Technical Specifications, Pg 22
Selected Administrative Procedures, BAP 1400-6, Pg 49
BAP 1400-6, Tech Spec Limiting Conditions for Operation Action Requirement LCOAR) procedure, Pg 8 43(b)(4)

Explanation:
This is as stated on Page 8 of BAP 1400-6. The LOSF is referred to as the SFD (Safety function Determination). This section states that If any supported system LCO entries are precluded by 3.0.6 based on a review of the BOL, these supported systems shall be documented on the DEL to ensure any future inoperabilities are properly evaluated on the SFD.
A. is correct as described above.
B. is incorrect for this information. It can be noted in the log, but required in the DEL.
C. is incorrect since this is required to be documented separately as a reminder and not duplicate LCOAR generation.
D. is incorrect for basically the same reason as C. above.

Date Written: 3/24/2006 Author: G. Wolfe App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
96	SRO	3		194001	2.2.3	3.1	3.3	Low

System/Evolution Name: Generic

Category Statement: Equipment Control

KA Statement:
(multi-unit) Knowledge of the design, procedural, and operational differences between units.

UserID:

Topic

Question Stem:

Compare the expected response between Unit 1 and Unit 2 for establishing feedwater flow to a Dry Steam Generator after initiation of Feed and Bleed as part of Attachment B in 1/2BFR-H.1, Response to Loss of Secondary Heat Sink.

A Dry SG is any SG with Wide Range level < 10% (27% ADVERSE CNMT) on ___(1)___ and,

if feedwater flow has been stopped for > 75 minutes with SG level < 45% Narrow Range, the feed line is considered voided on ____(2)____, and

the feedwater initial flowrate is determined by whether the tempering line is at least 75°F subcooled on ___(3)____.

	(1)	(2)	(3)
A	both Units	Unit 2	Unit 1
B	Unit 1	Unit1	Unit 1
C	both Units	Unit 1	Unit 2
D	Unit 2	Unit 2	Unit 2

Answer: C

Task No:

Obj No: T.FR3-04-A/E

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
1/2BFR-H.1, Attachment B, Response to Loss of Secondary Heat Sink, Pg 53
I1-FR-XL-03, FRPs-BFR H.1-H.5, Pg 54 - 56
43(b) (1)

Explanation:
With the different types of SGs in the two Units, Unit 1 SGs have a large, single feedline with a loop seal. Unit 2 has a very small upper feedline that accomodates ~10% of full flow at 100% power and all flow when the Unit is shutdown. Dispite the different SGs, both use the same Dry SG level criteria. Unit 1 is concerned about voiding in the feed ring and engineering evaluation determined that after 75 minutes without feed and NR SG level < 45%, the feedline may be void (full of steam), so this became the criteria for reducing feed flow when recomencing feed flow to prevent damage to the feed ring. Unit 2 has a much smaller feedline and the concern is thermal shocking of the feed nozzle and is precluded by throttling feed flow if the tempering line is not > 75°F subcooled. These differences are proceduralized to prevent damage to either Unit by applying one criteria to the other.
A. is incorrect since the application of voiding is Unit 1and subcooling is Unit 2.
B. is incorrect since the application of voiding is Unit 1but subcooling is Unit 2.
C. is correct as described above.
D. is incorrect since the application of voiding is Unit 1.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
97	SRO	3		194001	2.2.3	2.1	3.1	High

System/Evolution Name:
Generic

Category Statement:
Radiological Controls

KA Statement:
Knowledge of the requirements for reviewing and approving release permits.

UserID: Topic

Question Stem:

A gas release from Containment is pending on Unit 2 at 100% power using Containment Mini Purge System with the following progression:

- 0200 on 6/19 - Containment sample analyzed.
  - 2RE-PR011B, CNMT ATMOS UNIT 2, is in service.
- 0500 on 6/19 - Release package requested from Radiation Protection (RP).
- 1200 on 6/19 - Release was approved by Lead RP.
- 1500 on 6/19 - Release was approved by the shift SRO.

What is the expiration time/date for this release package per BCP 400-TCNMTROUTINE, Gaseous Effluent Release Form?

A 0000 on 6/20

B 0200 on 6/20

C 0800 on 6/20

D 1500 on 6/20

Answer: C

Task No: GW-001, S-HP-002,

Obj No:

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
BCP 400-TCNMTROUTINE, Gaseous Effluent Release Form
Byron SRO Certification Guide (OJT), Pg 22, 33
43(b) (6)

Explanation:
A NOTE on the first page of the release form states that analyzed samples are only good for 30 hours provided 2RE-PR011B has remained stable. The SRO approval records the expiration time/date based on 30 hours after the sample was obtained by RP.
A. is incorrect but plausible since some release durations are only good for the current day.
B. is incorrect but plausible since this would be a 24 hour duration from sampling, which is a common sample frequency when a release monitor is OOS.
C. is correct as described above.
D. is incorrect but plausible since this would be 24 hours from SRO approval, which is a common sample frequency for a release monitor being OOS.

Date Written: 3/25/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
98	SRO	3		194001	2.3.10	2.9	3.3	Low

System/Evolution Name: Generic

Category Statement: Radiological Controls

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

UserID: Topic

Question Stem:

Unit 2 is in MODE 6 with refueling activities in progress when the following occurs:

- A fuel assembly is dropped during removal from the core.
- Bubbling is observed from the core region.

What is your FIRST required action as the SRO in CNMT for Fuel Handling operations?

A Direct operations to establish CNMT closure.

B Direct the Control Room to start CNMT Charcoal Filter units.

C Direct the Control Room to announce, "ALL personnel to evacuate Unit 2 CNMT".

D Direct the Fuel Handlers to PLACE any fuel assembly in the transfer device into the change fixture.

Answer: C

Task No:

Obj No: S.OA29-03

Question Source: Byron NRC exam bank (1994)

Question Difficulty: Medium

Time: 1

Cross Ref:

Reference:
2BOA REFUEL-1, Fuel Handling Emergency. Pg 1, 2 entry conditions and step 1
I1-OA-XL-29, BOA REFUEL-1, Fuel Handling Emergency, Pg 4
43(b)4),(7)

Explanation:
This is actually a Radiological Fundamental and is the first step in the BOA.
A. is incorrect since this would have been established prior to moving fuel.
B. is incorrect since this is not the first action, but will be considered as followup action.
C. is correct as described above.
D. is incorrect since this is not an option immediately, and additonal movement must be evaluated first.

Date Written: 3/11/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
99	SRO	3		194001	2.4.34	3.8	3.6	High

System/Evolution Name: Generic

Category Statement: Emergency Procedures/Plan

KA Statement:
Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.

UserID: Topic

Question Stem: nit 1 control room has been evacuated and 1BOA PRI-5, Control Room Inaccessability, has been implemented due to a fire in the Upper Cable Spreading Room.

The Remote Shutdown Panel (RSP) is being activated.

- The NSO has aligned the RSP and 1PL05JA per Attachment A of 1BOA PRI-5.
- The NSO reports he has NO indication of 1D SG level or pressure at the RSP.

What action(s) is/are directed for this report?

A If MSIVs are open, use any other SG for pressure, balance feedwater flow to be consistent with the other 3 SGs.

B If MSIVs are closed, stop feeding the 1D SG, stop the 1D RCP, and verify 1D SG PORV is closed.

C Dispatch an operator to the Unit 1 Fire Hazards Panel to establish communications with the NSO at the RSP and align 1D SG level and pressure indication.

D Verify 1D SG MSIV is closed, isolate AF Flow to 1D SG, initiate a cooldown to < 550°F with the other 3 SGs and verify 1D SG PORV remains closed.

Answer: C

Task No:

Obj No: T.OA16-03/05/08

Question Source: New

Question Difficulty: Medium

Time: 1

Cross Ref:

1BOA PRI-5, Control Room Inaccessability, step 8

Reference:
I1-OA-XL-16, BOA PRI-5, Control Room Inaccessability, Pgs 16
43(b) (5) RMC

Explanation:
Although the suggested alternatives seem plausible, the proper direction in the BOA is to dispatch an operator to the Fire Hazards Panel to place only the required indications in local that can not be obtained at the RSP. Only 1A and 1D SG parameters are available at the Fire Hazards Communications will need to be established since the controls for the SG are still at the RSP.
A. is incorrect, although a plausible alternative, not the procedural action.
B. is incorrect, again, plausible, not procedural.
C. is correct as directed in the procedure.
D. is incorrect, again plausible, not procedural.

Date Written: 3/27/2006 Author: M. Jorgensen App. Ref:

Quest No:	RO SRO:	TIER:	GROUP:	Topic No:	KA No:	RO:	SRO:	Cog Level:
100	SRO	3		194001	2.4.36	2.0	2.8	Low

System/Evolution Name:
Generic

Category Statement:
Emergency Procedures/Plan

KA Statement:
Knowledge of chemistry / health physics tasks during emergency operations.

UserID: Topic

Question Stem:

Unit 2 Reactor has been manually tripped due to a steamline break on the 2B SG in the Safety Valve room.

The crew has transitioned to 2BEP-2, Faulted Steam Generator Isolation, and is currently performing Step 6, Check Secondary Radiation.

What direction is given to the Chemistry Department prior to exiting 2BEP-2?

A Sample ALL SGs for activity.

B Sample ALL INTACT SGs for activity.

C Continuously sample the FAULTED SG for activity.

D Sample the RCS for boron concentration

Answer: A

Task No:

Obj No: T.EP3-03

Question Source: New

Question Difficulty: Low

Time: 1

Cross Ref:

2BEP-2, Faulted Steam Generator Isolation, Pg 9

Reference:
I1-EP-XL-03, BEP-2, Faulted Steam Generator Isolation, Pg 9, 10
43(b)(4)

Explanation:

This is an important direction since a SGTR could also be in progress or could occur with a substantial delta-p across the tube sheet in the faulted SG with it depressurized and also provide information important for transition to the best recovery procedure. If no activity has caused an alarm the transition will be to 2BEP-1. If activity exists then a transition to 2BEP-3 will be required. So, it is equally important to sample ALL SGs for activity.

A. is correct as described above and in 2BEP-2.

B. is incorrect since all are equally important and, with 2B open to the environment, this one could be even more likely to develop high activity.

C. is incorrect since this would not allow sampling the 3 SGs that are being used for plant temperature control and may be steaming to atmosphere.

D. is incorrect since this sample will be directed the appropriate procedure and is not the concern in 2BEP-2.

Date Written: 3/26/2006 Author: M. Jorgensen App. Ref: