INITIAL SUBMITTAL OF THE RO/SRO WRITTEN EXAMINATION

FOR THE CLINTON INITIAL EXAMINATION - JULY 2005

Question #	1

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	203000 K2.02	2.5	2.7	Lower
System/Evolution Name:				**Category:**		
RHR/LPCI: Injection Mode				Plant Systems		
KA Statement:						
Knowledge of electrical power supplies to the following: Valves						

Which ONE of the following describes the impact of a loss of 480V Unit Sub 1B?

A. 1E12-F042C, LPCI From RH C Shutoff Valve, will NOT open electrically.

B. The ONLY source of AC power to the RPS Solenoid Bus B Inverter will be via the Bypass Transformer.

C. The SUCTION side of RWCU will NOT automatically isolate if Standby Liquid Control is initiated.

D. VC Train 'B' will operate ONLY in the High Radiation Isolation Mode.

Answer: A

Explanation:

A is correct – Per LP85205, Attachment D, the power supply to this MOV is AB MCC 1B4. Per CPS 3514.01C006, Section 2.1.1, the loss of 480V Unit Sub 1B results in the loss of all of the listed MCC's, including AB MCC 1B4. Without 480VAC motor power, the F042C valve will not open electrically.

B is incorrect – Per LP85434, Figure 7, only a single 480V bus supplies both the normal power to the inverter's rectifier section, and the alternate (backup) power through the bypass transformer. Additionally, this RPS Solenoid Bus B uses non-vital power; 480 V Unit Sub 1B is vital power.

C is incorrect – Per LP85204, page 35, a SLC Pump 'A' start signal closes the 1G33-F004 valve (RWCU Suction Outboard Isolation), while a SLC Pump 'B' start closes the 1G33-F001 valve (RWCU Suction Inboard Isolation). Per CPS 3514.01C006, Appendix A, page 46, a loss of 480V Unit Sub 1B disables 1G33-F001 (Suction Inboard) and 1G33-F040 (Return Inboard). When operators initiate SLC (start both pumps), the 'A' SLC Pump start will still close 1G33-F004, although the 'B' SLC Pump start will NOT close 1G33-F001. Because F004 does automatically close, the 'suction side' of RWCU does automatically isolate.

D is incorrect – Per CPS 3514.01C006, Appendix A, page 41, a loss of 480V Unit Sub 1B entirely disables VC (Control Rom HVAC) Train 'B' (a Div 2 subsystem). Only VC Train 'A' (not 'B') remains available, and only in the High Rad Isolation Mode (because of the failed-high PRMs, 1RIX-PR009B/D).

Objective:	Question Source:	Level of Difficulty:
LP85205.1.12	New	3.1

References provided to examinee:	None
References:	LP85205, Residual Heat Removal (RHR) LP85204, Reactor Water Cleanup (RWCU) LP85434, Nuclear System Protection System (NSPS (Inverters)) CPS 3514.01C006, 4160V Bus 1B1 Outage

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	2

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	209001 K2.01	3.0	3.1	Lower

System/Evolution Name:	Category:
Low Pressure Core Spray System	Plant Systems

KA Statement:
Knowledge of electrical power supplies to the following: Pump power

Which ONE of the following describes the impact of a loss of DC MCC 1A?

A. DG 1A CANNOT be started from the main control room, but CAN be started from its LOCAL CONTROL PANEL.

B. Reactor Recirculation Pump 'A' will automatically trip IF it is running in SLOW speed.

C. A running SF Pump will automatically trip IF the Suction Outboard Isolation Valve, 1SF004, closes.

D. LPCS Pump CANNOT be started from the main control room, but CAN be started at 4160V Bus 1A1.

Answer: D

Explanation:

D is correct – Per CPS 3514.01C040, Appendix A, page 31, loss of this bus results in the loss of LPCS Pump breaker control power. Since this pump has no DC control power-dependent starting interlocks, operators can still start the pump by locally closing the pump motor breaker at 4160V Bus 1A1.

A is incorrect – Per CPS 3514.01C040, Appendix A, page 31, all control power for this diesel is lost, both local and remote.

B is incorrect – Per CPS 3514.01C040, Appendix A, page 32, the 3A breaker for RR Pump 'A' loses its breaker logic control power. Per CPS 5003-5F, the 5A breaker will trip if the 3A breaker loses a control power source. The 5A breaker is closed only when the RR Pump is running in FAST speed.

C is incorrect – Per CPS 3514.01C040, Appendix A, page 32, a running SF Pump will NOT trip (trip interlock is disabled) if the suction valve, 1SF004, leaves it open seat.

Objective:	Question Source:	Level of Difficulty:
LP85209.1.12.3	New	3.2

References provided to examinee:	None
References:	LP85209, Low Pressure Core Spray, LPCS CPS 3514.01C040, 125 VDC MCC 1A Div 1 Outage CPS 5003-5F, RPT BKR LOSS OF DC CONT PWR

Date Written:	02/04/05	Author:	Ryder
Comments: None			

Question #	3

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	209002 K3.03	3.9	4.1	Lower

System/Evolution Name:	Category:
High Pressure Core Spray System	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) will have on the following: Adequate Core Cooling

Among the following ECCS SYSTEM FAILURES, which one would have the MOST SEVERE *impact on the ability to cool the core?*

A. With the reactor initially at rated power, a SMALL-break LOCA occurs and the HPCS Pump will NOT run.

B. With the reactor initially at rated power, a LARGE-break LOCA depressurizes the plant and the LPCS Pump will NOT run.

C. After a scram from rated power, reactor water level drops to TAF, and NONE of the ADS Valves will open.

D. After a scram from rated power, reactor water level is declared UNKNOWN, and NONE of the ADS Valves will open.

Answer: A

Explanation:
A is correct – Per LP85380, pages 4, 5 and 7, the HPCS system failure (when compared to the other answer choices) would have the most severe impact on the ability to cool the core, for both small and large-break LOCAs. The lesson plan discussion is very brief and does not attempt to qualify (describes the analyses) why this is the case. Although the USAR (see attached references) does provide this clarification, such information is beyond scope for the application of this KA to the RO Exam. Therefore, the correct answer has been carefully stated to be consistent with both the lesson plan claim, and the USAR discussion. Similarly, each distracter has been stated in a way that is certain to <u>not</u> conflict with any of the assumptions belonging to the USAR analyses, while at the same time providing sufficient plausibility. B is incorrect - For the reasons described above. C and D *are incorrect* – For the reasons described above. Also, the claim here is that the ADS Valves (an ECCS system) fail to open. There are 9 other non-ADS SRVs available to be opened.

Objective:	Question Source:	Level of Difficulty:
None	New	3.9

References provided to examinee:	None
References:	LP85380, High Pressure Core Spray, HPCS USAR, Section 6.3.3, ECCS Performance Evaluation

Date Written:	02/04/05	Author:	Ryder
Comments: None			

Question #	4

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	211000 K1.03	2.5	2.6	Lower

System/Evolution Name:	Category:
Standby Liquid Control System	Plant Systems

KA Statement:
Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant air systems

With the plant operating at rated power, the service air SPARGE valve for the SLC Storage Tank has been unintentionally left OPEN.

WITHOUT operator action, which ONE of the following describes the EARLIEST potential impact on the SLC system, as a result of this mispositioned sparge valve?

A. Burn out of the SLC Storage Tank OPERATING Heater

B. HIGHER Boron concentration in the SLC Storage Tank

C. SLC Storage Tank overflow through the top vent

D. LOWER Boron concentration in the SLC Storage Tank

Answer: B

Explanation:
B is correct – This question is written directly from CPS LER 2004-002-00 (see attached references). The continuous sparge resulted in tank water evaporation and a rise in boron concentration as a consequence. This LER is included in the lesson plan, LP85211, Attachment C, (OPEX) discussion. A is incorrect – But is quite plausible; so plausible, that the 'EARLIEST' component of the question stem is critical to avoiding a second correct answer. Per LP85211, page 20, if tank level lowers to <1,000 gallons remaining, the Operating Heater could be damaged due to being uncovered. However, since normal SLC tank level is about 4,000 gallons (LP85211, page 6), uncovering the heater would NOT be the 'earliest' potential impact. C and D are incorrect – For the reasons associated with the correct answer.

Objective:	Question Source:	Level of Difficulty:
LP85211.1.12.2	New	2.3

References provided to examinee:	None
References:	LP85211, Standby Liquid Control (SLC) CPS LER 2004-002-00, Mispositioned SLC Air Sparge Valve Results in High Boron Concentration

Date Written:	04/28/05	Author:	Ryder
Comments: None			

Question #	5

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	212000 A1.07	3.4	3.4	Lower

System/Evolution Name:	Category:
Reactor Protection System	Plant Systems

KA Statement:
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: Rod position information

BEFORE a scram is RESET, which ONE of the following describes an ACCEPTABLE method to determine that a given control rod HAS FULLY inserted?

A. Confirm the ROD FULL IN light is lit, for that rod, at either RACC panel.

B. With the Full Core Display in DUAL mode, confirm that EITHER channel's Full-In LED is lit AND the numerical display is Blank (NO numerical value).

C. With the Full Core Display in 2 CHANNEL mode, confirm that BOTH channels' Full-In LEDs are lit AND the numerical display indicates '00'.

D. Confirm a value of 0 (zero) volts on Transient Test Channel 291.

Answer: B

Explanation:

B is correct – Per CPS 4100.01, Section 2.1.2, as described in this choice.

A is incorrect – Per CPS 4100.01, Section 2.1.2, and CPS 3304.02, Section 8.2.11.2. There is only single 'All Rods Full In' LED at these RACC panels. There is no individual 'Rod Full In' light for each given rod. This choice is very plausible to the Candidate who vaguely recalls that there is a way to manually address each given rod's actual position (including 00), using the ID Generator, at these RACC panels.

C is incorrect – This choice suggests a variation of the correct answer, 'B', but it is not correct (see the correct answer's explanation).

D is incorrect – Per CPS 3304.02, Section 8.2.11.3. A zero (0) volts value is associated with 'all rods not full in.'

Objective:	Question Source:	Level of Difficulty:
DB410001.1.6	New	2.7

References provided to examinee:	None
References:	LP85401, Rod Control & Information System DB410001, Reactor Scram CPS 4100.01, Reactor Scram CPS 3304.02, Rod Control and Information System

Date Written:	05/03/05	Author:	Ryder
Comments: None			

Question #	6

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	215004 K6.01	3.2	3.3	Higher

System/Evolution Name:	Category:
Source Range Monitor (SRM) System	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: RPS

The plant is operating at rated power when SRM 'A' fails UPSCALE.

Which ONE of the following describes the plant response if ONLY THE DIV 1 shorting link were loose and had become dislodged (circuit interrupted) before the SRM failed?

A. Scram, ONLY

B. Rod block, ONLY

C. Rod block AND scram

D. NEITHER a scram, NOR a rod block

Answer: A

Explanation:

A is correct – Per CPS 5005-1K, rod block is bypassed with the Mode Switch in RUN ('plant operating at rated power'). Per CPS 5005-1K, scram function is dependent on neither the Mode Switch position, nor IRM Range. It solely depends on shorting link status. Per LP85212, Figure 13, all it takes is the removal of a single shorting link (for that Division) to enable the non-coincident scram function. SRM 'A' belongs to Div 1 RPS.

B, C, and D are incorrect – For the reasons described above. 'D' is quite plausible to candidate who does recognize that the rod block is already bypassed (RUN), but has never considered the specific impact of having ONLY a single shorting link removed (rather than ALL of them being removed, as would be the case for special testing that might require such a non-coincident scram function). It is also plausible to the candidate who believes that, like the rod block, the SRM scram function is also bypassed in RUN.

Objective:	Question Source:	Level of Difficulty:
LP85215.1.4.5	New	2.7

References provided to examinee:	None
References:	LP85212, Reactor Protection System LP85215, SRMs CPS 5005-1K, SRM UPSC ALARM OR INOP

Date Written:	02/06/05	Author:	Ryder
Comments: None			

Question #	7

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	215005 K1.07	2.6	2.9	Higher

System/Evolution Name:	Category:
Average Power Range Monitor/Local Power Range Monitor	Plant Systems

KA Statement:
Knowledge of the physical connections and/or cause-effect relationships between APRM/LPRM and the following: Process computer, performance monitoring system

ASSUME the following when answering this question:

- The neutron flux levels seen by each of the LPRMs inputting to APRM 'A' are IDENTICAL

The plant is operating at power, with the following:

- 3D-Monicore calculated Core Thermal Power (CTP) is 90%
- APRM 'A' is reading 90%
- The 'As Found' AGAF reading (3D-Monicore) for APRM 'A' is 1.000

- THEN, a single LPRM inputting to APRM 'A' fails to a ZERO value signal

- The failed LPRM has NOT yet been bypassed

Which ONE of the following describes the RESULTING AGAF reading for APRM 'A'?

The AGAF is reading...

A. Lower than 0.980.

B. 0.980 to 0.999.

C. 1.001 to 1.020.

D. Higher than 1.020.

Answer: D

Explanation:

D is correct – Reference LP85411 and CPS 9431.60, throughout this discussion. AGAF calculation: AGAF = % CTP ÷ APRM reading.

- **Conditions prior to the LPRM failure:**
    - 90% APRM reading resulting from 33 <u>equal</u> LPRM flux signals
- **Conditions post-LPRM failure ('bad' LPRM still feeding the APRM):**
    - (32 ÷33) x 90% = 87.3% APRM reading
      AGAF is now reading: AGAF = 90% + 87.3% = **1.031**

A, B, and C are incorrect – For the reasons described above.

Question #	7

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	215005 K1.07	2.6	2.9	Higher

System/Evolution Name:	Category:
Average Power Range Monitor/Local Power Range Monitor	Plant Systems

KA Statement:
Knowledge of the physical connections and/or cause-effect relationships between APRM/LPRM and the following: Process computer, performance monitoring system

Objective:	Question Source:	Level of Difficulty:
LP85211.1.13.3; .1.15.1	New	3.2

References provided to examinee:	None
References:	LP85411, APRM/LPRM System CPS 9431.60, APRM Gain Adjustment

Date Written:	02/07/05	Author:	Ryder

Comments:

Operational Validity basis for this question: 1) Operators must understand the meaning of the AGAF value, rather than just recognize whether an 'as found' value is SAT or UNSAT (per Tech Spec SR 3.3.1.1.2); 2) operators must recognize that an 'as found' value >1.000 is non-conservative, where a value <1.000 is relatively conservative (albeit still a potential Tech Spec concern); and 3) operators must recognize that even a single LPRM failure (hard upscale or hard downscale) can inop the APRM, for sake of the surveillance requirement (3.3.1.1.2). Depending on current rod pattern, total rod density, and fuel distribution, the failed-to-zero condition of an LPRM that was already producing a much weaker flux signal (as compared to the other 32 that feed the given APRM), may not affect the aggregate signal enough to push the AGAF value above the 1.020 upper limit; this is the reason for the stem's opening 'ASSUME' statement.

Question #	8

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	217000 K4.06	3.5	3.5	Higher

System/Evolution Name:	Category:
Reactor Core Isolation Cooling (RCIC)	Plant Systems

KA Statement:
Knowledge of the REACTOR CORE ISOLATION COOLING (RCIC) design features and/or interlocks which provide for the following: Manual initiation

Operators are ready to MANUALLY initiate RCIC from its normal standby lineup.

Which ONE of the following explains why the RCIC Manual Initiate pushbutton MUST be HELD DEPRESSED FOR 6 SECONDS?

To allow enough time for...

A. RCIC Pump Supply to Turbine Lube Oil Cooler Valve, 1E51-F046, to fully open, enabling the opening circuit for RCIC Steam Supply Valve, 1E51-F045.

B. a logic time delay device to energize, enabling the opening circuit for RCIC Steam Supply Valve, 1E51-F045.

C. the Ramp Generator to BEGIN its ramping period.

D. the Ramp Generator to FINISH its ramping period.

Answer: B

Explanation:
B is correct – Per LP85217, page 44, and drawing E02-1RI99, Sheets 6 and 9. The 'TD' device shown on Sheet 9 must be energized (timed out) before the initiation signal can enable the F045 opening circuit. CPS 9054.03, Section 8.2.2.2, validates that this TD device is calibrated for about 6 seconds. Only after holding the pushbutton depressed for about 6 seconds does F045 begin to open. Direction on how to manually initiate RCIC is found in CPS 3310.01, Section 8.1.3. A is incorrect – Refer to LP85217, pages 44-47, for the explanation related to all of the distracters. There is NO electrical connection between the open limit switch for F046 and the opening circuit for F045. C and D are incorrect – The Ramp Generator does not even come into the picture until 6 to 9 seconds <u>after</u> the F045 valves <u>begins</u> to open. Refer to LP85217, page 45.

Objective:	Question Source:	Level of Difficulty:
LP85217.1.15.7	New	3.0

References provided to examinee:	None
References:	LP85217, Reactor Core Isolation Cooling CPS Drawing E02-1RI99, Sheets 6 and 9, RCIC Schematic Diagram CPS 9054.03, RCIC Simulated Auto Actuation Test CPS 3310.01, RCIC

Date Written:	04/28/05	Author:	Ryder
Comments: None			

Question #	9

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	239002 A1.06	3.7	3.8	Higher

System/Evolution Name:	Category:
Relief/Safety Valves	Plant Systems

KA Statement:
Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including: Reactor power

With the plant operating at 90% power, a Safety Relief Valve (SRV) INADVERTENTLY OPENS.

Which ONE of the following predicts how a plant parameter INITIALLY responds when the SRV opens, and describes the reason why?

INITIALLY, indicated reactor...

A. water level LOWERS, because of the RPV inventory lost through the open SRV.

B. power LOWERS, because the SRV opening causes a slight drop in reactor pressure.

C. water level RISES, because Feedwater Level Control immediately sees the additional steam flow.

D. power RISES, because of the reduced feedwater inlet temperature.

Answer: B

Explanation:
B is correct – Per USAR, Section 15.1.4.3.3, the SRV opening initially produces a slight depressurization transient. Per Generic Fundamentals knowledge, the drop in reactor pressure produces more voiding, which initially lowers reactor power (see LP85756S, page 26, for an analogous 'off-normal' pressure transient, which validates the relationship between pressure and power). A and C are incorrect – Per Generic Fundamentals knowledge, as well as the USAR discussion above, the initial depressurization causes more voiding, which results in an initial RISE of reactor water level ('swell' transient). The open SRV diverts main steam flow away from (is upstream of) the main steam line flow element (see LP85239, Figure 1). This results in the Feedwater Level Control System immediately seeing a lower steam flow, not a higher steam flow. D is incorrect – Per CPS 4005.01, Loss of Feedwater Heating, the SRV opening is a loss of feedwater heating event. The escape of steam to the suppression pool diverts it away from the main turbine and the extraction steam supply. Reactor feedwater temperature lowers (i.e., a greater amount of core inlet sub-cooling is produced) and this positive reactivity addition should help to raise reactor power. However, this is not the INITIAL power response. The depressurization event immediately lowers reactor power.

Objective:	Question Source:	Level of Difficulty:
None	New	3.6

Question #	10

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	259002 K5.07	2.7	2.7	Lower

System/Evolution Name:	Category:
Reactor Water Level Control System	Plant Systems

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: Turbine speed control mechanisms: TDRFP

The plant is operating at rated power with Feedwater and Feedwater Level Control in their NORMAL configurations.

Per CPS 3103.01, Feedwater, which ONE of the following describes the expected CURRENT Feedwater Level Control operating configuration, <u>and</u> the reason for that configuration?

A. TDRFP Manual Speed Potentiometers are set at the LOW SPEED STOP position, to expedite taking manual control of a locked up TDRFP.

B. RFPT Flow Controllers are in MANUAL, to expedite the Emergency Restart of a tripped TDRFP.

C. TDRFP Manual Speed Potentiometers are set at the ZERO speed position, to expedite the Emergency Restart of a tripped TDRFP.

D. RFPT AUTO/MAN XFER switches are in MANUAL, to expedite taking manual control of a locked up TDRFP.

Answer: C

Explanation:
C is correct – Per CPS 3103.01, Sections 2.1.7, 8.1.4, and 8.3.2. The Manual Speed Pot must be at zero speed (fully CCW) in order to reset the control logic for the LP and HP control valves. This ensures that when the operator RESETS the TDRFP, immediate control of speed will be available. During a normal plant and feedwater system startup, the last time that operators have reason to manipulate the Manual Speed Pot is in Section 8.1.4.20(1)(e). It is here that the pot is set to ZERO speed position and should remain there with 'Feedwater and Feedwater Level Control in their normal configurations'. A is incorrect – For the reasons described above. B is incorrect – Per CPS 3103.01, Section 8.1.8, the TDRFPs are being controlled by the Master Level Controller, which means that each RFPT Flow Controller is in AUTOMATIC. D is incorrect – Per LP85570, page 24, and Figure 13, these XFER switches are in AUTO (pushbutton depressed) whenever any flow controller is controlling the TDRFP.

Objective:	Question Source:	Level of Difficulty:
LP85570.1.14	New	2.2

References provided to examinee:	None
References:	LP85570, Feedwater Level Control System CPS 3103.01, Feedwater

Question #	11

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	261000 K4.01	3.7	3.8	Higher
System/Evolution Name:				**Category:**		
Standby Gas Treatment System				Plant Systems		
KA Statement:						
Knowledge of STANDBY GAS TREATMENT SYSTEM design features and/or interlocks which provide for the following: *Automatic system initiation*						

The plant is in MODE 4, with the following:

- BOTH trains of Standby Gas Treatment (VG) are in a STANDBY lineup
- The ENTIRE Div 1 NSPS Bus is in an OUTAGE

- THEN, the CNMT Bldg Exhaust Radiation Monitor, 1RIX-PR001C, fails UPSCALE and produces a trip

WITHOUT operator action, which ONE of the following identifies the VG Trains that are RUNNING, and explains why?

A. BOTH, because with the Div 1 NSPS Bus outage, the failure of 1RIX-PR001C completes the 'one-out-of-two-twice' initiation logic.

B. NEITHER, because no VG initiation signal is present.

C. ONLY Train 'A', because no Train 'B' initiation signal is present.

D. ONLY Train 'B', because with the Div 1 NSPS Bus outage, VG Train 'A' auto-initiation signals are disabled.

Answer: B

Explanation:
B is correct – Refer to CPS 5140.61 for the one-out-of-two-twice radiation monitor combinations that can produce a VG initiation signal. Although both trains are capable of auto-starting, a trip condition on a single radiation monitor channel (1RIX-PR001C, only) does not satisfy the one-out-two-twice initiation logic. A is incorrect – The Div 1 NSPS Bus outage does not produce a power failure trip of any of the 1RIX-PR001 channels. Per LP85273, page 74 (Attachment B), these radiation monitors get their power from Auxiliary Power System MCC's, not from NSPS (inverter) power. Therefore, the upscale trip produced by the PR001C failure does not, alone, satisfy the one-out-of-two-twice VG initiation logic. C is incorrect – This is plausible to the candidate who believes the NSPS Bus outage does produce a power failure trip on the PR001A channel, who confuses the one-out-of-two-twice logic combinations that produce a VG initiation signal, and who mistakenly associates two of the PR001 channels with VG Train 'A' and the other two channels with VG Train 'B'. This 'ladder logic' is a fairly common weakness among candidates who have not mastered this system knowledge. D is incorrect – This is quite plausible to the candidate who believes the NSPS Bus outage does produce a power failure trip on the PR001A channel, and recalls that (per CPS 3509.01C001, Appendix A, page 25) the Bus outage disables the auto-start of VG Train 'A' on a LOCA signal (High DW Pressure, Low-Low Level, only). The radiation monitor initiation signals are not affected by this Bus outage.

Question #	12

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	262001 A2.01	3.4	3.6	Higher

System/Evolution Name:	Category:
A.C. Electrical Distribution	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine/generator trip

Reactor power is 25% when the following occurs:

- Main turbine and generator trip (cause unknown)
- ALL LOADS on 6900V Bus 1B become DE-ENERGIZED
- There are NO indications of electrical faults on any buses or breakers

Which ONE of the following describes how the operator restores power to 6900V Bus 1B?

A. Place the Sync Switch to ON for the UAT 1B feeder breaker, position the UAT 1B feeder breaker control switch to CLOSE, then place the Sync Switch to OFF.

B. Verify the RAT source is dead, then position the UAT 1B feeder breaker control switch to CLOSE.

C. Place the Sync Switch to ON for the RAT feeder breaker, position the RAT feeder breaker control switch to CLOSE, then place the Sync Switch to OFF.

D. Verify the UAT 1B source is dead, then position the RAT feeder breaker control switch to CLOSE.

Answer: C

Explanation:
C is correct – Refer to LP85571, Figure 4, and page 31, and to CPS 3501.01, Section 8.1.1. The normal feed for 6.8 KV Bus 1B is from UAT 1B. When the generator trips, UAT 1B (and 1A) are lost, forcing the auto-transfer of 6.9 KV Bus 1B to the RAT (its reserve feed). Because the bus is 'dead', procedure section 8.1.1 is used to complete the transfer manually. Steps 8.1.4, 5, and 7 are featured in this answer choice.
A and B are incorrect – The UATs are lost on the generator trip. These choices are distracting to the Candidate who cannot recall the 'Main' (UAT 1B) versus 'Reserve' (RAT) power sources for this 6900V bus.
D is incorrect – Whether the bus is alive or dead is irrelevant. The Sync Switch must still be used.

Objective:	Question Source:	Level of Difficulty:
LP85571.1.4.6	New	2.5

Question #	12

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	262001 A2.01	3.4	3.6	Higher

System/Evolution Name:	Category:
A.C. Electrical Distribution	Plant Systems

KA Statement:

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine/generator trip

References provided to examinee:	None
References:	LP85571, Auxiliary Power CPS 3501.01, High Voltage Auxiliary Power System

Date Written:	04/28/05	Author:	Ryder

Comments:

The way this question addresses the 'predict' component of this KA deserves some mention. The 2nd bullet of the stem conditions demands that the candidate: 1) determine if this is an expected condition, and 2) if not, where should the loads have transferred to?. Having 'predicted' the desired condition for loads on this bus, the candidate proceeds directly to 4 choices that demand that he/she correct the undesired condition. He/she can choose the correct answer only by knowing where the loads should have auto-transferred to.

Question #	13

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	262001 A4.05	3.3	3.3	Higher
System/Evolution Name:			**Category:**			
A.C. Electrical Distribution			Plant Systems			
KA Statement:						
Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses						

After transferring loads to the ERAT, operators are preparing to shut down the RAT SVC from the main control room, in accordance with CPS 3505.03, RAT & ERAT Static VAR Compensators.

EXISTING readings for the RAT SVC, at panel P870, are as follows:

- 4,220 Volts
- - 4 MVARs

Which ONE of the following identifies the FINAL voltage value that the SVC Voltmeter should ramp to AFTER the operator places the RAT SVC control switch to OFF?

A. 4,060 Volts

B. 4,140 Volts

C. 4,300 Volts

D. 4,380 Volts

Answer: C

Explanation:
C is correct – Refer to CPS 3505.03, Section 8.3 and Appendix A. The - MVARs indication means that the SVC is acting to hold down voltage. When the SVC is removed from service, the resulting (uncompensated) voltage will ramp up (to above 4,220 volts). The rule of thumb is 20 volts per MVAR. In this case, that amounts to +80 volts, or a FINAL value of 4,300 volts. A, B, and D are incorrect – For the reasons described. They have face validity and are plausible to the candidate who either, does not recall the rule of thumb (applies 40 volts per MVAR instead of 20 volts per MVAR), or cannot distinguish between a + MVAR reading and a – MVAR reading.

Objective:	Question Source:	Level of Difficulty:
LP85305.1.10.2	New	2.6

References provided to examinee:	None
References:	LP85305, Static VAR Compensator CPS 3505.03, RAT & ERAT Static VAR Compensators

Date Written:	02/09/05	Author:	Ryder
Comments: None			

Question #	14

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	262002 K4.01	3.1	3.4	Lower

System/Evolution Name:	Category:
Uninterruptable Power Supply (A.C./D.C.)	Plant Systems

KA Statement:
Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies

Which ONE of the following describes the impact of a loss of the NORMAL power supply to a DIVISIONAL NSPS Inverter Cabinet, with the Cabinet in its NORMAL operating configuration?

A. The bus loads will REMAIN ENERGIZED as a 125 VDC bus automatically begins to feed the Inverter section.

B. The bus loads will REMAIN ENERGIZED as a Static Switch automatically transfers them to an alternate 120 VAC supply.

C. The bus loads will BECOME DE-ENERGIZED and remain that way until operators MANUALLY transfer them using the REVERSE TRANSFER pushbutton.

D. The bus loads will BECOME DE-ENERGIZED and remain that way until operators MANUALLY transfer them using the TRANSFER SWITCH.

Answer: B

Explanation:

B is correct – Refer to LP85434, pages 11, 13, 14, and Figures 2b and 3. A Divisional NSPS Cabinet is normally supplied from the associated 125 VDC Bus. With the Cabinet in its normal operating configuration, the Transfer Switch is in the INVERTER position. This allows the Static Switch to auto-transfer bus loads to the alternate 120 VAC supply.

A is incorrect – This describes the impact of a non-Divisional NSPS Inverter Cabinet. See LP85434, Figure 6.

C is incorrect – Operators use this pushbutton to manually transfer bus loads to the alternate 120 VAC supply per CPS 3509.01, Section 8.1.4.

D is incorrect – For the reasons associated with the correct answer.

Objective:	Question Source:	Level of Difficulty:
LP85434.1.4.3	CPS Operations Exam Bank, Question #10278 (DIRECT, editorial changes only)	2.7

References provided to examinee:	None
References:	LP85434, Nuclear System Protection System (NSPS) CPS 3509.01, Instrument Power System

Date Written:	02/10/05	Author:	Ryder
Comments: None			

Question #	15

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	263000 A2.02	2.6	2.9	Higher

System/Evolution Name:	Category:
D.C. Electrical Distribution	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging

The following statement describes a **FACT** concerning battery hydrogen production:

- The RATE at which a battery produces hydrogen during an EQUALIZING charge is DIRECTLY proportional to the battery capacity (in Ampere-Hours).

Consider the above FACT when answering the following question.

An EQUALIZING Charge of the Div 3 Battery is in progress, when normal battery room ventilation (VX) is lost.

Which ONE of the following:

(1) predicts the RATE at which Div 3 Battery Room hydrogen concentration will rise, WITHOUT operator action, AFTER normal battery room ventilation is lost,
<u>and</u>
(2) describes the required action?

The hydrogen concentration RATE OF RISE will be...

A. (1) GREATER in the EARLY hours of the Equalizing Charge.
(2) IF room hydrogen concentration reaches 2%, THEN open the battery room door and ventilate with a portable air blower.

B. (1) GREATER in the LATER hours of the Equalizing Charge.
(2) IF room hydrogen concentration reaches 2%, THEN open the battery room door and ventilate with a portable air blower.

C. (1) GREATER in the EARLY hours of the Equalizing Charge.
(2) Open the battery room door; WHEN hydrogen concentration reaches 5%, THEN ventilate the room with a portable air blower.

D. (1) GREATER in the LATER hours of the Equalizing Charge.
(2) Open the battery room door; WHEN hydrogen concentration reaches 5%, THEN ventilate the room with a portable air blower.

Answer: B

Explanation:
B is correct – Concerning Part (1) of the question, refer to information extracted from the Web source: <u>www.dep.state.pa.us/dep/deputate/minres/dms/website/training/battery.pdf</u>. *This is battery training program* presentation associated with the U.S. Bureau of Mines. These slides show that the rate of hydrogen production ('H') is <u>directly</u> proportional to the capacity of the battery (in ampere-hours). Since the battery capacity rises over the charging

Question #	15

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	263000 A2.02	2.6	2.9	Higher

System/Evolution Name:	Category:
D.C. Electrical Distribution	*Plant Systems*

KA Statement:
Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging

period, so too does the rate of gas production ('H'). Concerning Part (2) of the question, refer to CPS 3412.01, Section 8.2.3.

A, C, and D are incorrect – *For the reasons described above.*

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

References provided to examinee:	None
References:	CPS 3412.01, Essential Switchgear Heat Removal (VX) www.dep.state.pa.us/dep/deputate/minres/dms/website/training/battery.pdf

Date Written:	05/02/05	Author:	Ryder

Comments:

This question is categorized as Higher Cognitive (HCL), because the Candidate must 'associate' the given stem claim regarding the rate of hydrogen production with his/her understanding of how a battery's capacity changes over the period of being re-charged. It is also HCL because, as a closed-reference question, in order for the Candidate to eliminate choices 'C' and 'D', he/she must recognize the danger that would exist if a portable blower (i.e., a potential spark producing device) were to be started when hydrogen is already above a combustible concentration (nominally 4-5%).

Question #	16

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	264000 A2.01	3.5	3.6	Higher
System/Evolution Name:				**Category:**		
Emergency Generators (Diesel/Jet)				Plant Systems		
KA Statement:						
Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Parallel operation of emergency generator						

The plant is operating at rated power, with the Monthly surveillance for DG 1B in progress, with the following:

- DG 1B is running loaded at 3,800 KW
- THEN, the NORMAL control signal to the Woodward Governor is lost
- CRS determines the need to CORRECT the DG 1B operating condition that has resulted from this Governor malfunction

Which ONE of the following:

(1) predicts the response of DG 1B to the governor control signal failure,
and
(2) describes the action necessary to correct the DG's current operating condition?

A. (1) Engine speed REMAINS THE SAME, but Load RISES.
(2) Emergency STOP the DG from the main control room.

B. (1) Engine Speed REMAINS THE SAME, but Load LOWERS.
(2) RAISE the SETPOINT for the Mechanical Governor, locally.

C. (1) Engine Speed RISES, but Load REMAINS THE SAME.
(2) LOWER the engine speed using the Governor control switch.

D. (1) Engine Speed LOWERS, but Load REMAINS THE SAME
(2) RAISE the engine speed using the Governor control switch.

Answer: A

Explanation:
A is correct – Refer to CPS 9080.02, page 28 CAUTION. The monthly surveillance has the DG loaded in parallel with the off-site power. Per LP 85264, pages 15-16, the electrical governor is 'normally' controlling the engine speed, and when the electrical signal is lost (fails low), the mechanical governor assumes control at a 5% HIGHER governor setpoint. Because the DG is paralleled with off-site, engine speed cannot change, but the DG does pick up more load. Although we cannot necessarily predict that it will pick up 5% additional load (for a new load of 3,990 KW), it will certainly pick up an amount of load that causes the DG to operate in excess of its 'continuous rating of 3,875 KW' (see Section 6.2.11 of CPS 9080.02). This question suggests that it is this 'operating condition' that needs to be corrected. Once the CRS decides to 'correct' the condition, the only way to do so is by completely unloading the DG. With the failed electric governor, and with NO procedure guidance that would allow operators to manually lower the mechanical governor's setpoint (locally), the required action is to Emergency STOP the machine. B is incorrect – For the reasons described above.

Question #	16

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	264000 A2.01	3.5	3.6	Higher

System/Evolution Name:	Category:
Emergency Generators (Diesel/Jet)	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Parallel operation of emergency generator

C and D are incorrect – These choices suggest the 'predicted' response of the DG if it were not paralleled with off-site. The Candidate is expected to know that the machine is paralleled with off-site when running loaded for the Monthly surveillance.

Objective:	Question Source:	Level of Difficulty:
None	New	3.5

References provided to examinee:	None
References:	*LP85264, Diesel Generator/Diesel Fuel Oil* CPS 9080.02, Diesel Generator 1B Operability (i.e., the Monthly)

Date Written:	04/15/05	Author:	Ryder

Comments:

This question is an RO/SRO one, and is not an SRO-ONLY question, for the following reason:

1. It may appear, at first, that the question is presented in a way that is consistent with other 'A2' type exam questions that have been categorized as SRO-ONLY, but a closer look shows that it is quite different.

2. The last stem condition bullet has pre-empted the need for the SRO to make a decision about whether to correct the DG operating condition, or not.

3. The suggested choices for the Part (2) 'required action' are not a set of choices from which only the SRO would be expected choose. Rather, each choice challenges the Candidate (both RO/SRO) to recognizing what is the only possible/permitted way of correcting the operating condition.

4. Therefore, this question is presented in a way that really amounts to requiring only several pieces of 'systems' type of knowledge: 1) the configuration that the DG is in before the governor failure (i.e., paralleled with off-site), and on the electric governor; 2) how the DG engine speed and load respond as a result of the 5% off-set between the electric and mechanical setpoints; and 3) recognition of the fact that with there being no electric governor control, the only solution to emergency STOP the DG.

5. As such, this is a question that should be on both the RO and SRO Exams.

Question #	17

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	400000 K6.06	2.9	2.9	Higher
System/Evolution Name:				**Category:**		
Component Cooling Water System (CCWS)				Plant Systems		
KA Statement:						
Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Heat exchangers and condensers						

The plant is in MODE 5 for a refueling outage, with the following:

- Shutdown Service Water (SX) Pumps A and C are out-of-service and have Clearances installed
- THEN, a complete loss of Service Water (WS) occurs
- SX Pump B starts and runs
- An electrical failure PREVENTS the associated WS to SX Header Isolation Valve from FULLY closing
- NLO reports that SX Header Pressure reads 90 psig, locally
- THEN, a tube leak occurs in the CCW Heat Exchanger (HX) that was operating before the loss of WS

Which ONE of the following describes the POTENTIAL consequence of the CCW HX tube leak?

A. Reduced heat transfer due to heat transfer surface fouling

B. Higher rate of depletion of the CCW Demineralizer resin

C. Radioactive discharge to the environment

D. Rising level in the CCW Expansion Tank

Answer: C

Explanation:
C is correct – Per LP85208, page 29. SX system pressure is lower than CCW system pressure. CCW inventory will be lost through the tube leak and find its way into the Lake. A, B, and D are incorrect – Per LP85208, page 29. These are all indicative of a WS-to-CCW leak (WS at higher pressure than CCW).

Objective:	Question Source:	Level of Difficulty:
LP85208.1.10.1	New	2.8

References provided to examinee:	None
References:	LP85208, Component Cooling Water System

Date Written:	05/02/05	**Author:**	Ryder
Comments: None			

Question #	18

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	204000 K3.06	2.6	2.7	Higher

System/Evolution Name:	Category:
Reactor Water Cleanup System	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on the following: Area radiation levels

Following a maintenance outage on RWCU Filter 'A', with the plant operating at rated power, the following conditions exist:

- RWCU Filter System Functions Interlock Switch is in the 'SYS A' position
- RWCU Filter 'A' has just been manually re-filled in preparation for Backwash
- Backwash Receiving Tank (BWRT) level is reading 38% (local panel)
- Operators are unaware that BWRT level is reading about 25% LOWER than ACTUAL level in the Tank

Which ONE of the following describes the POTENTIAL consequence associated with the NEXT operator action related to the Backwash of RWCU Filter 'A'?

A. RWCU system isolation on High Differential Flow

B. Higher than normal area radiation level on CNMT el. 778'

C. RWCU system isolation on Equipment Room High Temperature

D. Higher than normal area radiation level on Auxiliary Bldg el. 737'

Answer: B

Explanation:
B is correct – Per CPS 3303.02, Sections 8.7.6 and 8.6.5, the NEXT operator action is to return the System Function Interlock Switch to the NORM(al) position. Candidates need not recall (from memory) such a procedural action as this; rather, they must only recognize that the filter backwash requires the System Functions Interlock Switch be in NORMAL. Per LP85204, page 29, this can result in overflowing the Backwash Receiving Tank (BWRT) through a continuous vent connected to the CNMT building HVAC exhaust ductwork. This will cause significant contamination and elevated area radiation levels throughout the CNMT spaces. A and C are incorrect – The Filter is still in the 'Shutdown' Mode (see CPS 3303.02, Sections 2.2.8 and 8.6). As such, the Filter is still isolated from the RWCU system and system isolations are not possible. D is incorrect – This is the location of the RWCU Pumps. There is no physical, or ventilation air-flow, connection between an existing high area radiation level in the CNMT building and the Auxiliary Building. And, there is no system perturbation being suggested by the stem conditions that could cause a RWCU Pump problem (e.g., a seal leak) that would result in high radiation levels in that pump area.

Objective:	Question Source:	Level of Difficulty:
LP85204.1.15	New	3.5

Question #	18

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	204000 K3.06	2.6	2.7	Higher

System/Evolution Name:	Category:
Reactor Water Cleanup System	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on the following: Area radiation levels

References provided to examinee:	None
References:	LP85204, Reactor Water Cleanup System CPS 3303.01, Reactor Water Cleanup System CPS 3303.02, RWCU Filter Demineralizer Operating Procedure

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	19

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	201005 K5.10	3.2	3.3	Higher
System/Evolution Name:				**Category:**		
Rod Control and Information System (RCIS)				Plant Systems		
KA Statement:						
Knowledge of the operational implications of the following concepts as they relate to ROD CONTROL AND INFORMATION SYSTEM (RCIS): Rod withdrawal limiter						

Which ONE of the following describes a situation where the Technical Specifications ALLOW (permit, without administrative restrictions) ALL normal control rod movements (In and Out) to be performed?

Reactor Power is...

A. 40%; the light above the LO POWER SET PT is OFF, and the light above the LO POWER ALM PT is OFF.

B. 10%; the light above the LO POWER SET PT is ON.

C. 75%; the light above the HI POWER SET PT is OFF.

D. 45%; the light above the LO POWER SET PT is OFF, and the light above the LO POWER ALM PT is ON.

Answer: D

Explanation:

D is correct – Refer to LP85401, pages 23-24, and Figure 4, to Tech Spec 3.3.2.1, and to CPS 3005.01, Section 6.2, for all of the answer choices. Reactor power is within the range when the RWL must be OPERABLE (>29% RTP and at or below the High Power Setpoint (HPSP) of 70% RTP). The fact that the LO POWER SET PT light is OFF, while the LO POWER ALM PT light is ON, indicates a properly-functioning Low Power Function of the RWL, permitting any type of rod movement (subject to the built-in notch restraints of the RWL itself). There are NO Tech Spec administrative restrictions with these conditions.

A is incorrect – Reactor power is within the range when the RWL must be OPERABLE (>29% RTP and at or below the High Power Setpoint (HPSP) of 70%). However, the fact that both of these lights are OFF indicates that the Low Power Function of the RWL (normally enabled by the Rod Pattern Controller, RPC) is in fact bypassed, making the RWL INOPERABLE. Per TS LCO 3.3.2.1.A, all control rod WITHDRAWALS must be immediately suspended. Insertions are still allowed.

B is incorrect – Reactor power is below the Low Power Setpoint (LPSP) of 16.7% RTP. However, the fact that the light is ON indicates that the RPC is INOPERABLE. Per Tech Spec LCO 3.3.2.1.B, all normal rod movements (in and out) must be immediately suspended.

C is incorrect – Reactor power is above the High Power Setpoint (HPSP) of 70% RTP. However, this light being OFF indicates that the High Power Function of the RWL is bypassed, making the RWL INOPERABLE. Per Tech Spec LCO 3.3.2.1.A, all control rod WITHDRAWALS must be immediately suspended. Insertions are still allowed.

Objective:	Question Source:	Level of Difficulty:
None	New	2.7

Question #	19

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	201005 K5.10	3.2	3.3	Higher

System/Evolution Name:	Category:
Rod Control and Information System (RCIS)	Plant Systems

KA Statement:
Knowledge of the operational implications of the following concepts as they relate to ROD CONTROL AND INFORMATION SYSTEM (RCIS): Rod withdrawal limiter

References provided to examinee:	None
References:	LP85401, Rod Control and Information System CPS Tech Spec 3.3.2.1, Control Rod Block Instrumentation CPS 9436.05, RPC Low Power Setpoint Channel Calibration CPS 9030.01C021, RPC Low Power Setpoint Checklist CPS 3005.01, Unit Power Changes

Date Written:	05/04/05	Author:	Ryder

Comments:

This question is on the RO Exam (as opposed to be classified as SRO-ONLY) because the RO Candidate needs only to understand the meaning of each of these lights (systems knowledge), recognize the impact of a given light's status on the RWL/RPC operability, and then recall (from memory) the applicable <1-hour Tech Spec Actions (i.e., to 'immediately suspend' rod movements).

Question #	20

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	202001 A4.01	3.7	3.7	Lower

System/Evolution Name:	Category:
Recirculation System	Plant Systems

KA Statement:
Ability to manually operate and/or monitor in the control room: Recirculation pumps

The Reactor Recirc Pumps are being manually transferred from FAST to SLOW speed in accordance with CPS 3302.01, Reactor Recirculation.

Which ONE of the following describes ONE of the operator actions involved in performing this transfer?

In the main control room, the operator...

A. manually closes the LFMG Motor Breakers, CB-1A(B).

B. positions the FCVs to about 19% open BEFORE the transfer.

C. verifies the RECIRC MG A(B) INTERLOCK BYPASS annunciators are extinguished.

D. positions the FCVs to about 76% open AFTER the transfer.

Answer: A

Explanation:
A *is correct* – Per CPS 3302.01, Section 8.1.3. Although the transfer sequence logic would automatically close CB-1A and B (see LP85202, page 28, and Figure 7), the operating procedure directs the operator to manually close these breakers before initiating the transfer sequence. B and D are incorrect – Per CPS 3302.01, Sections 8.1.3.4 and 8.1.3.7. About 10% open BEFORE the transfer, and about 90% open AFTER the transfer. C is incorrect – Per CPS 3302.01, Section 8.1.3.1, and CPS 5003-4C. These interlocks are intentionally bypassed for this evolution, causing these annunciators to be in alarm (<u>not</u> extinguished).

Objective:	Question Source:	Level of Difficulty:
None	New	2.8

References provided to examinee:	None
References:	LP85202, Reactor Recirculation System CPS 3302.01, Reactor Recirculation System CPS 5003-4C, Recirc MG A Interlock Bypass

Date Written:	02/17/05	Author:	Ryder
Comments: None			

Question #	21

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	214000 A3.01	3.4	3.3	Lower

System/Evolution Name:	Category:
Rod Position Information System	Plant Systems

KA Statement:
Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including: Full Core Display

The RO is performing a control rod coupling check per CPS 3304.02, Rod Control and Information System.

WHILE a continuous withdrawal signal is being applied, which ONE of the following indicates that the control rod is UNCOUPLED?

A. CRD drive water flow reads 5 gpm.

B. ROD OVERTRAVEL annunciator is NOT received.

C. Red 'full-out' light is LIT on the full-core display.

D. Rod position is BLANK on the full-core display.

Answer: D

Explanation:
D is correct – Per CPS 3304.02, Section 8.2.6 NOTE, rod position would be blank on the RIDM (full-core display) for an uncoupled rod. A is incorrect – Per Section 8.1.10 NOTE. Whether CRDM seals are good (1-3 gpm stall flow indicated), or bad (something higher than 1-3 gpm), stall flow is unaffected by the status of the control rod blade (coupled, or uncoupled). This is the reason why stall flow is to be used ONLY as an indication of seal condition. B is incorrect – Per CPS 3304.02, Section 8.2.6 NOTE, the Rod Overtravel annunciator <u>would</u> be received for an uncoupled rod. C is incorrect – Per CPS 3304.02, Section 8.1.10.1, 2nd bullet.

Objective:	Question Source:	Level of Difficulty:
LP85401.1.4.9	New	2.8

References provided to examinee:	None
References:	CPS 3304.02, Rod Control & Information System

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	22

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	233000 A2.07	3.0	3.2	Higher

System/Evolution Name:	Category:
Fuel Pool Cooling and Cleanup	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEANUP: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: *High fuel pool temperature*

Which ONE of the following:

(1) describes a POTENTIAL or ACTUAL concern associated with a Spent Fuel Storage Pool Temperature that has risen to 152°F and has STABILIZED there,
<u>and</u>
(2) describes the operational impact?

A. (1) Exceeds the ORM OPERATING REQUIREMENT for Spent Fuel Storage Pool temperature.
(2) Movement of fuel assemblies in the pool is NOT permitted.

B. (1) Results in elevated humidity in the Fuel Building.
(2) If the reactor is operating, a normal plant shutdown is required.

C. (1) Exceeds the TECHNICAL SPECIFICATION LCO for Spent Fuel Storage Pool temperature.
(2) Movement of fuel assemblies in the pool is NOT permitted.

D. (1) Results in airborne radioactivity in the Fuel Building.
(2) If the reactor is shutdown, it must remain shutdown.

Answer: D

Explanation:
D is correct – Part (1), per LP85233, page 45. Part (2), per CPS 3317.01, Section 4.6. A and C are incorrect – There is no Tech Spec LCO, or ORM OR, related to Spent Fuel Storage Pool Temperature. B is incorrect – Although Part (1) is correct, there is no procedural requirement for shutting down the plant. See the attached CPS 5040-1F and CPS 3317.01, Section 8.2.4, to validate this claim.

Objective:	Question Source:	Level of Difficulty:
LP85233.1.14	New	3.4

References provided to examinee:	None
References:	LP85233, Fuel Pool Cooling and Cleanup CPS 3317.01, Fuel Pool Cooling and Cleanup CPS 5040-1F, High Temp Spent Fuel Storage Pool

Question #	22

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	233000 A2.07	3.0	3.2	Higher

System/Evolution Name:	Category:
Fuel Pool Cooling and Cleanup	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the FUEL POOL COOLING AND CLEANUP: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High fuel pool temperature

Date Written:	04/29/05	Author:	Ryder

Comments: None

This question is presented on the RO Exam (and is not considered an SRO-ONLY type) because the 'operational impact' portion requires only the recall of an operating procedure Precaution/Limitation; it does not require any operational decision-making (reserved for the SRO's responsibility), nor does it require any form of 'application' (which might or might not be reserved for the SRO) of the information contained in that Precaution/Limitation.

Question #	23

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	241000 K6.01	2.8	2.9	Higher
System/Evolution Name:				**Category:**		
Reactor/Turbine Pressure Regulating System				Plant Systems		
KA Statement:						
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM: A.C. electrical power						

A main turbine roll-up is in progress, with the following:

- Control Building MCC 'C' is de-energized and has clearances installed
- THEN, FIVE MINUTES AFTER the RO depresses the '1800' pushbutton for main turbine Speed Set RPM, UPS Bus 1B is lost

Which ONE of the following describes the plant response?

The main turbine...

A. STOPS rolling up and STABILIZES at its CURRENT speed.

B. TRIPS, and the turbine bypass valves fail OPEN.

C. TRIPS, and the turbine bypass valves fail SHUT.

D. RETURNS to 100 RPM and STABILIZES there.

Answer: C

Explanation:
C is correct – Per LP85576, page 15, and CPS 3509.01C006, page 28. With main turbine speed at <75% of rated speed (.75 x 1800 rated rpm = 1350 rpm), the main turbine trips due to de-energization of the 24 VDC Trip Bus and Electrical Trip Solenoids. Even if the FAST 'Starting Rate' has been selected (see LP85241, page 30, and CPS 3105.01, Section *8.1.6), the machine will be running at NO HIGHER than about 900 RPM, at 5 minutes after depressing the 1800 RPM* pushbutton. In fact, per CPS 3105.01, Section 8.1.7.4 NOTE, it can take 3-4 minutes just see <u>any</u> speed increase on the machine. In this case, at the 5-minute mark, operators shouldn't expect to see the machine speed any higher than about 200-400 rpm. <u>Any</u> time all power is lost to the TBV control circuits, the TBVs will fail shut. NOTE: The stem condition regarding CB MCC 'C' being de-energized ensures the intended failure response of the TBVs; i.e., there <u>may</u> be an auctioneering of power (between this MCC and UPS 1B) to the TBV circuits. Taking this MCC away, ensures the TBVs <u>will</u> fail shut. A, B, and D are incorrect - For the reasons described above.

Objective:	Question Source:	Level of Difficulty:
LP85576.1.13.5	New	3.6

References provided to examinee:	None
References:	LP85241, Steam Bypass and Pressure Control System LP85576, Computer UPS CPS 3509.01C006, UPS 1B Bus Outage CPS 3105.01, Turbine

Question #	23

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	241000 K6.01	2.8	2.9	Higher

System/Evolution Name:	Category:
Reactor/Turbine Pressure Regulating System	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM: A.C. electrical power

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	24

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	259001 K2.01	3.3	3.3	Lower

System/Evolution Name:	Category:
Reactor Feedwater System	Plant Systems

KA Statement:
Knowledge of electrical power supplies to the following: Reactor feedwater pump(s): Motor-Driven only

Which ONE of the following describes the impact of a loss of 6.9KV Bus 1B?

A. Reactor Recirc Pump 1B can be started ONLY in SLOW speed.

B. Motor-Driven Reactor Feedwater Pump (MDRFP) will NOT run.

C. ONLY ONE Circulating Water Pump will run.

D. NEITHER Isolated Phase Bus Duct Cooler Fan will run.

Answer: B

Explanation:

B is correct – Per LP85259, page 12. MDRFP is powered from 6.9KV Bus 1B.

A is incorrect – Per LP85202, pages 9 and 27. Even a SLOW speed start of the RR Pump 1B requires 6.9KV Bus 1B power.

C is incorrect – Per LP85275, pages 14 and 17. 6.9KV Bus 1B powers only one of the 3 CW Pumps (CWP 'B'). The other two powered from 6.9KV Bus 1A. There are no inter-pump starting permissives or trip signals that would inhibit the running of both pumps, CWP 'A' and 'C', on the 6.9KV Bus 1A.

D is incorrect – Per LP85572, pages 8 and 13, and CPS drawing E02-1AP03 (and LP85571, Figure 4 for clarity). One of these fans is powered from 6.9KV Bus 1A, via 480V Unit Sub 1J. Only the 'B' Fan is lost if 6.9KV Bus 1B is lost.

Objective:	Question Source:	Level of Difficulty:
LP85259.1.4.4	New	2.0

References provided to examinee:	None
References:	LP85259, Feedwater System LP85202, Reactor Recirculation System LP85275, Circulating Water System LP85572, Isolated Phase Bus Duct Cooling

Date Written:	02/21/05	Author:	Ryder
Comments: None			

Question #	25

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	286000 K4.07	3.3	3.3	Higher
System/Evolution Name:			**Category:**			
Fire Protection System			Plant Systems			
KA Statement:						
Knowledge of FIRE PROTECTION SYSTEM design features and/or interlocks which provide for the following: Diesel engine protection						

Operators are testing the automatic start feature of the 'B' Fire Pump. The operator places the Mode Selector Switch in TEST, and the following occurs:

- At Time = 0 minutes, the engine begins to crank
- At Time = 4 minutes, the engine starts and runs
- At Time = 5 minutes, the engine stabilizes at 130% of rated speed
- At Time = 6 minutes, both a HIGH ENGINE TEMPERATURE alarm, and a LOW LUBE OIL PRESSURE alarm, are received on the XL3 fire alarm panels
- 30 seconds later, the operator manually stops the engine by placing the Mode Selector Switch to OFF

Which ONE of the following identifies the TOTAL NUMBER of AUTOMATIC engine protective features (i.e., should have prevented the engine from running) that FAILED during this test of the 'B' Fire Pump?

A. 1

B. 2

C. 3

D. 4

Answer: B

Explanation:
B is correct – Per LP85286, pages 21-25. One engine 'crank and rest' cycle takes 30 seconds; the controller should have allowed no more than 6 total cycles (180 seconds…At Time = 3 minutes) before stopping the auto-start sequence and generating a 'Failure to Start' alarm. This was the first failure of a protective action. The engine was allowed to reach 130% of rated speed and continue to run. The engine should have tripped (stopped) at 120% of rated speed. This was the second failure of a protective action. A is incorrect – For the reasons described above. C and D are incorrect – Per the same reference cited above. Neither the High Engine Temperature alarm, nor the Low Lube Oil Pressure alarm, provide an automatic protective action; they are alarms, only.

Objective:	Question Source:	Level of Difficulty:
LP85286.1.10.4	New	4.0

Question #	25

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	286000 K4.07	3.3	3.3	Higher

System/Evolution Name:	**Category:**
Fire Protection System	Plant Systems
KA Statement:	
Knowledge of FIRE PROTECTION SYSTEM design features and/or interlocks which provide for the following: Diesel engine protection	

References provided to examinee:	None
References:	LP85286, Fire Protection and Detection CPS 9071.02, Diesel Fire Pump Capacity Checks/Sequential Starting

Date Written:	02/21/05	**Author:**	Ryder
Comments: None			

Question #	26

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	288000 K1.05	3.3	3.6	Lower

System/Evolution Name:	Category:
Plant Ventilation Systems	Plant Systems

KA Statement:
Knowledge of the physical connections and/or cause-effect relationships between PLANT VENTILATION SYSTEMS and the following: Process radiation monitoring system

Which ONE of the following identifies the TOTAL NUMBER of HIGH RADIATION Isolation CONDITIONS that:

(1) can cause an isolation of the normal Continuous Containment Purge (CCP) lineup, and

(2) are overridden (considering ALL plant ventilation systems) if both Containment HVAC Isolation Valve Radiation Interlock Bypass Switches are placed in TOTAL BYPASS?

A. (1) Two
(2) Three

B. (1) Two
(2) Four

C. (1) Three
(2) Four

D. (1) Three
(5) Five

Answer: C

Explanation:
C is correct – Per LP85455, pages 4, 6, 49, 50, 51, and 52. Considering Part (1)…a total of THREE signals ('Conditions') will isolate CCP (Group 10 valves); they are: Containment Bldg Exhaust Radiation High; Containment Bldg Fuel Transfer Pool Vent Plenum Radiation High; and Containment CCP Exhaust Radiation High. Considering Part (2)…a total of FOUR signals ('Conditions') are overridden by placing these switches in TOTAL BYPASS; they are: the same THREE that isolate CCP, plus the Fuel Building Exhaust Radiation High signal (which does NOT close the CCP valves). A, B, and D are incorrect – For the reasons described above, but are plausible for any candidate who cannot recall the specific radiation signals that interface with CCP and the Interlock Bypass Switches.

Objective:	Question Source:	Level of Difficulty:
LP85455.1.4.11	New	3.4

References provided to examinee:	None
References:	LP85455, Containment Ventilation and Drywell Purge

Question #	26

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	288000 K1.05	3.3	3.6	Lower
System/Evolution Name:				**Category:**		
Plant Ventilation Systems				Plant Systems		
KA Statement:						
Knowledge of the physical connections and/or cause-effect relationships between PLANT VENTILATION SYSTEMS and the following: Process radiation monitoring system						

Date Written:	04/29/05	Author:	Ryder

Comments:

This question is categorized as Lower Cognitive (LCL) because, although a two-part question, there is NO cause-effect relationship between the first and second part; there is no required association, one with the other. Each part demands only one mental process from the Candidate: Part (1) – from memory, recall how many different radiation signals will isolate CCP; Part (2) – from memory, recall how many different radiation signals are overridden by the Total Bypass switch.

Question #	27

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.1.3	3.0	3.4	Higher

System/Evolution Name:	Category:
	Conduct of Operations

KA Statement:
Knowledge of shift turnover practices

Consider the following:

- You are the on-coming RO-A for dayshift, July 30
- You last stood an entire (8-hour) RO watch on dayshift, July 24

Per OP-AA-112-101, Shift Turnover and Relief, which ONE of the following identifies how far back, in time, you are required to review the Narrative Log before relieving the watch?

Back though, at least,....

A. the beginning of SWING shift on July 24.

B. 0000 hours on July 25.

C. the beginning of DAY shift on July 25.

D. 0000 hours on July 26.

Answer: D

Explanation:
D is correct – Per Section 4.8.3 of OP-AA-112-101. The on-coming RO is required to review the logs 'through the last previous date on shift', or 'the preceding four days logs',...'whichever is less'. The 'preceding four days' limit applies in this case. The four-day period that precedes July 30 begins at 0000 hours, July 26. A is incorrect – For the reasons described above. This choice would be correct if the procedure read...'whichever is <u>more</u>'. B and C are incorrect – For the reasons associated with the correct answer. These choices presume the candidate incorrectly recalls a 'preceding five days logs' requirement. Additionally, theses choices cause the candidate to ponder the meaning of 'preceding five days'.

Objective:	Question Source:	Level of Difficulty:
PBAD012, Objective 5	New	2.6

References provided to examinee:	None
References:	OP-AA-112-101, Shift Relief and Turnover

Date Written:	02/22/05	**Author:**	Ryder
Comments: None			

Question #	28

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.2.28	2.6	3.5	Higher

System/Evolution Name:	Category:
	Equipment Control

KA Statement:
Knowledge of new and spent fuel movement procedures

A refueling outage is in progress, when THREE of the SRM Channels fall below 3 counts per second.

Per CPS 3703.01, Core Alterations, which ONE of the following work evolutions CAN be performed?

A. Removal of an IRRADIATED control rod blade from a FULLY DE-FUELED cell in the core, and its transfer to the Spent Fuel Storage Pool.

B. Removal of an IRRADIATED fuel bundle from the core, and its transfer to the Spent Fuel Storage Pool.

C. Transfer of a NEW fuel bundle from the Upper Containment Storage Pool, and its installation into the core.

D. Transfer of a NEW control rod blade from the Upper Containment Storage Pool, and its installation into a core cell containing ONLY ONE fuel bundle.

Answer: A

Explanation:
A is correct – Per CPS 3703.01, Section 6.10, the inoperable SRMs cited in the question stem (<u>no</u> matter which 3 SRM Channels), would require a halt to Core Alterations Per CPS 3703.01, Section 2.2.4, however, this answer choice does NOT describe an evolution that is considered a Core Alteration (i.e., the evolution meets the criteria of the 2nd exception ('b') in the CORE ALTERATION definition). Therefore, this work CAN be performed with 3 inoperable SRMs. NOTE: This claim has been verified against the SRM Tech Spec (3.3.1.2), as well (see that reference, attached). B, C, and D are incorrect – Each of these choices <u>is</u> a Core Alteration, as defined in Section 2.2.4. Therefore, these describe work that CANNOT be performed with 3 inoperable SRMs.

Objective:	Question Source:	Level of Difficulty:
LP86610.1.19	New	2.9

References provided to examinee:	None
References:	CPS 3703.01, Core Alterations CPS Tech Spec 3.3.1.2, *SRM Instrumentation*

Date Written:	02/22/05	Author:	Ryder

Comments:

This question is Higher Cognitive (HCL) for at least <u>two</u> reasons:

1. The candidate must associate the stem's three failed SRMs with the requirements of Section 6.10 of the

Question #	28

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.2.28	2.6	3.5	Higher

System/Evolution Name:	Category:
	Equipment Control

KA Statement:
Knowledge of new and spent fuel movement procedures

procedure.

2. The candidate must determine whether any given one of the answer choices is/is not a type of Core Alteration.

Question #	29

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.3.4	2.5	3.1	Lower

System/Evolution Name:	Category:
	Radiological Controls

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

A Site Area Emergency is in progress, when it is determined that an Emergency Exposure is required SOLELY for the purpose of PROTECTING an important piece of PLANT EQUIPMENT.

Which ONE of the following:

(1) identifies the exposure LIMIT (TEDE) for this emergency exposure,
<u>and</u>
(2) identifies the HIGHEST level of approval needed to authorize this exposure limit?

A. (1) 7 Rem
(2) Radiation Protection Management

B. (1) 10 Rem
(2) Station Emergency Director

C. (1) 15 Rem
(2) Radiation Protection Management

D. (1) 25 Rem
(2) Station Emergency Director

Answer: B

Explanation:
B is correct – Part (1): Per RP-AA-203, Section 4.5.3, Table 2. 10 Rem TEDE is the limit for solely protecting property. Part (2): Per EP-AA-113-F-02, <u>all</u> Emergency Exposures (no matter the specific limit) require Station Emergency Director authorization. A, C, and D are incorrect – For the reasons described above; but all have face validity and are plausible to an uncertain candidate.

Objective:	Question Source:	Level of Difficulty:
None	New	2.8

References provided to examinee:	None
References:	RP-AA-203, Exposure Control and Authorization EP-AA-113-F-02, Authorization for Emergency Exposure

Date Written:	02/23/05	Author:	Ryder
Comments: None			

Question #	30

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.4.7	3.1	3.8	Higher

System/Evolution Name:	Category:
	Emergency Procedures and Plan

KA Statement:
Knowledge of event based EOP mitigation strategies

Using the provided references, answer the following.

An ATWS and LOCA are in progress, with the following:

- The MSL/OG and IA Interlocks (RPV Level 1) have been defeated per CPS 4410.00C004
- THEN, main condenser vacuum is lost and CANNOT be restored

Which ONE of the following mitigation strategies IS ALLOWED by the EOPs?

Using the...

A. RFPTs to stabilize pressure

B. MSL Drains to depressurize to the Shutdown Cooling pressure interlock once Cold Shutdown Boron has been injected

C. Condenser as an Alternate Depressurization System in EOP-3, or as a way to reduce pressure below 50 psig in EOP-2

D. MSL Drains to stabilize pressure

Answer: C

Explanation:
C is correct – Per CPS 4411.09, Section 2.2 NOTE. Without a condenser vacuum, the EOPs allow the condenser to be used as a vent path (i.e., an alternate depressurization path), but not as a heat sink. Additionally, the stem conditions indicate that MSIVs (Group 1) and Turbine Bypass Valves have closed on the loss of vacuum (see LP85255, page 20). Therefore, only where an EOP step states that it is 'OK to defeat RPV vent interlocks' can operators defeat the low vacuum closure. Both EOP-2 and EOP-3 allow this. Stem conditions indicate that operators have already defeated the RPV Level 1 isolation for Group 1 (MSIVs) (see attached reference). A, B, and D are incorrect – For the reasons described above. These choices are suggest using the condenser as heat sink when there is no vacuum. These choices are all taken from the Pressure leg of EOP-1A.

Objective:	Question Source:	Level of Difficulty:
None	New	3.9

Question #	30

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.4.7	3.1	3.8	Higher
System/Evolution Name:				**Category:**		
				Emergency Procedures and Plan		
KA Statement:						
Knowledge of event based EOP mitigation strategies						

References provided to examinee:	EOP flowchart set
References:	EOP-1A, ATWS RPV Control EOP-2, RPV Flooding EOP-3, Emergency RPV Depressurization (Blowdown) CPS 4410.00C004, Defeating MSL/OG Interlocks CPS 4411.09, RPV Pressure Control Sources LP85255, Condenser Air Removal System

Date Written:	02/24/05	**Author:**	Ryder
Comments: None			

Question #	31

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.4.24	3.3	3.7	Lower

System/Evolution Name:	Category:
	Emergency Procedures and Plan

KA Statement:
Knowledge of loss of cooling water procedures

The plant is operating at rated power when a COMPLETE LOSS OF ALL suction capability occurs at the Screenhouse, coincident with a Station Blackout (SBO).

Five minutes later, the following conditions exist:

- The reactor is shutdown
- There is NO LOCA condition
- Reactor pressure and water level are STABLE in their normal bands

Which ONE of the following identifies the system/equipment considered 'most critical for worst case survivability'?

A. RCIC

B. DG 1C

C. HPCS

D. FC

Answer: A

Explanation:
A is correct – Per CPS 4303.01, Appendix A, Section 1, RCIC is the system considered 'most critical for worst case survivability'. B and C are incorrect – But either is plausible to the Candidate who recalls that, per CPS 4200.01, Section 1.4, HPCS is the preferred source of RPV makeup, given that DG 1C (Div 3 power) is assumed to be available during a SBO. D is incorrect – But provides sufficient plausibility for the Candidate who leans towards giving a greater priority to keeping fission products in solution in the Spent Fuel Pool Storage, given that the stem conditions indicate there is no RPV inventory control problem (i.e., no LOCA).

Objective:	Question Source:	Level of Difficulty:
None	New	3.1

References provided to examinee:	None
References:	CPS 4303.01, Loss of the Ultimate Heat Sink CPS 4200.01, Loss of AC Power

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	32

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	209002 A3.01	3.3	3.3	Higher
System/Evolution Name:				**Category:**		
High Pressure Core Spray System (HPCS)				Plant Systems		
KA Statement:						
Ability to monitor automatic operations of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) including: Valve operation						

The plant is operating at rated power, with the following:

- CPS 9051.01, HPCS Pump Operability (QUARTERLY surveillance), is in progress AT FULL RATED FLOW
- THEN, HPCS automatically initiates on a High Drywell Pressure signal

Which ONE of the following identifies HPCS system valves that will show an INTERMEDIATE position indication (on P601) IMMEDIATELY AFTER the initiation signal is received?

A. HPCS to CNMT Outboard Isolation Valve, 1E22-F004, and HPCS Test Valve to Suppression Pool, 1E22-F023

B. HPCS to CNMT Outboard Isolation Valve, 1E22-F004, and HPCS Second Test Valve to Storage Tank, 1E22-F011

C. HPCS Suction from RCIC Storage Tank Valve, 1E22-F001, and HPCS First Test Valve to Storage Tank, 1E22-F010

D. HPCS Suction from RCIC Storage Tank Valve, 1E22-F001, and HPCS Suppression Pool Suction Valve, 1E22-F015

Answer: B

Explanation:
B is correct – Per CPS 9051.01, Section 8.2, CPS 3309.01, Section 8.1.2, and LP85380, Figure 4. The flowpath for this surveillance is Storage Tank-to-Storage Tank (<u>not</u> Suppression Pool-to-Suppression Pool). Upon receipt of the HPCS auto-initiation signal, the following occur simultaneously (from this pre-initiation lineup): F004 (injection valve) begins to stroke open (shows intermediate position), and HPCS First and Second Test Valves to Storage Tank (F010 and F011) stroke closed (show intermediate position). A is incorrect – With the surveillance running using this flowpath, F023 is fully closed and remains that way throughout the initiation. C is incorrect – Although F010 begins to stroke closed, F001 is already fully open (for the surveillance) and remains that way throughout the initiation (until an automatic suction swap, if any, occurs later on). D is incorrect – As described for choice 'C', F001 is already open and remains that way. Interlocks prevent F001 and F015 from ever stroking at the same time.

Objective:	Question Source:	Level of Difficulty:
LP85380.1.8	New	2.6

Question #	32

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	209002 A3.01	3.3	3.3	Higher

System/Evolution Name:	Category:
High Pressure Core Spray System (HPCS)	Plant Systems

KA Statement:
Ability to monitor automatic operations of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) including: Valve operation

References provided to examinee:	None
References:	LP85380, High Pressure Core Spray CPS 3309.01, High Pressure Core Spray CPS 9051.01, HPCS Pump Operability

Date Written:	02/28/05	Author:	Ryder

Comments:

This question is categorized as Higher Cognitive (HCL) because the Candidate must associate the pre-LOCA lineup of HPCS (what does the 'Preferred' test lineup mean?) with the specific valves that are necessary to realign (out of the test lineup), post-LOCA.

Question #	33

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	218000 A3.08	4.2	4.3	Higher
System/Evolution Name:			**Category:**			
Automatic Depressurization System			Plant Systems			
KA Statement:						
Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Reactor pressure						

Which ONE of the following results in the FASTEST INITIAL RATE of reactor pressure reduction?

A. With reactor pressure initially at 900 psig, operators are unable to Inhibit ADS, and ALL of the ADS Valves automatically open.

B. With reactor pressure initially at 900 psig, operators open ALL of the Turbine Bypass Valves using the Bypass Jack.

C. Immediately after a turbine trip from 50% power, NO Turbine Bypass Valves open (cause unknown), and reactor pressure PEAKS at 1090 psig.

D. With the Pressure Regulator at its NORMAL setpoint, ALL of the LLS-SRVs have automatically reclosed, THEN reactor pressure rises and PEAKS at 970 psig.

Answer: A

Explanation:
A is correct – Per LP85239, Attachments F and G, and page 10. Each of the 16 SRVs are capable of relieving reactor pressure at the same rate (about 6.5% of total rated steam flow). There are 7 SRVs that open when operators initiate ADS (also found in LP85218, page 4). B is incorrect – Per CPS 3105.04, Section 2.1.1. Total relief capacity is 28.8%, for the 6 TBVs, or about 5% for each TBV (as compared to 6.5% for each SRV). Additionally, only 6 TBVs open, as compared to 7 SRVs (when ADS is initiated). C is incorrect – Per LP85239, Attachment G. SRV F051D has a pressure-relief setpoint low enough (1103 psig +/- 15 psig, per Tech Spec 3.3.6.5) to open the SRV if pressure peaks at only 1089 psig. As soon as it does, the Low-Low Set (LLS) circuits as activated for all 5 LLS-SRVs. However, only F051C has a LLS opening setpoint (1073 +/- 15 psig) low enough to open, shortly after F051D opened. This choice, therefore, proposes an event where no more than a total of 2 SRVs (and none of the 6 TBVs) open to effect a pressure reduction. D is incorrect – Per LP85241, page 6, the Pressure Regulator is normally set at 930 psig (steam pressure, not reactor pressure). All 5 LLS-SRVs automatically reclosing suggests reactor pressure has fallen below 957 psig. A rise, again, and a peaking at 970 psig re-opens NONE of these LLS-SRVs, and NONE of the 6 TBVs (i.e., steam pressure is still well below its 930 psig Pressure Regulator setpoint).

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

Question #	33

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	218000 A3.08	4.2	4.3	Higher

System/Evolution Name:	**Category:**
Automatic Depressurization System	Plant Systems

KA Statement:
Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Reactor pressure

References provided to examinee:	None
References:	LP85218, Automatic Depressurization System LP85239, Main Steam System LP85241, Steam Bypass and Pressure Control System CPS Tech Spec 3.3.6.5, Relief and LLS Instrumentation CPS 3105.04, Steam Bypass and Pressure Regulator

Date Written:	05/16/05	**Author:**	Ryder
Comments: None			

Question #	34

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	218000 2.4.31	3.3	3.4	Lower

System/Evolution Name:	Category:
Automatic Depressurization System	*Plant Systems*

KA Statement:
Knowledge of annunciator alarms and indications, and use of response instructions

Assuming the alarming condition is VALID, which ONE of the following annunciators, BY ITSELF, reminds the operators that an EOP entry condition ALREADY exists?

A. 5065-6F, SECONDARY CNMT AREA HIGH TEMP

B. 5066-5A, ADS LOGIC B 105 SEC TIMER INITIATED

C. 5064-7C, ECCS FLOOR DRAIN SUMP HIGH LEAK RATE

D. 5064-5C, SUPPR POOL DIVISION 1 HIGH TEMPERATURE

Answer: B

Explanation:

B is correct – Per CPS 5066-5A. This annunciator alarms only when a high drywell pressure (1.68 psig) and confirmed low-low-low water level (-145.5 inches) condition (or confirmed low-low-low level for >6 minutes) exists. These parameters represent EOP-1 and EOP-6 entry conditions. The procedure's "Operator Actions" remind the operators of this.

A is incorrect – Per CPS 5065-6F. This annunciator represents area high temperature alarm values, not the 'max normal' values associated with EOP-8 entry.

C is incorrect – Per CPS 5064-7C. This annunciator monitors the status of various ECCS room (in secondary containment) floor drain sump systems (i.e., high leakage resulting in excessive pump-down time and/or frequency. Although this leakage problem may progress to where floor water levels in those rooms require EOP-8 entry, water levels are not there yet. As such, no EOP entry yet exists.

D is incorrect – Per CPS 5064-5C. This annunciator represents a 90°F suppression pool temperature. The EOP-6 entry value is 95°F. As such, no EOP entry yet exists.

Objective:	Question Source:	Level of Difficulty:
LP85218.1.11.5	New	2.7

References provided to examinee:	None
References:	LP85218, Automatic Depressurization System CPS 5064-5C response procedure CPS 5064-7C response procedure CPS 5065-6F response procedure CPS 5066-5A response procedure

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	35

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	223002 K3.22	2.5	2.6	Higher

System/Evolution Name:	Category:
PCIS/Nuclear Steam Supply Shutoff System (NSSSS)	Plant Systems

KA Statement:
Knowledge of the effect that a loss or malfunction of the PCIS/NSSSS will have on the following: Containment drainage system

The plant is operating at rated power, when a DIV 2 NSPS circuit malfunction causes the INPUT to the Load Driver, that services Group 8 isolation valves, to fail to ZERO.

Which ONE of the following valves CLOSES as a result of this failure?

A. INBOARD Containment Equipment Drain Sump Discharge Valve, 1RE021

B. OUTBOARD Containment Equipment Drain Sump Discharge Valve, 1RE022

C. OUTBOARD Containment Building Supply Isolation Valve, 1VR001A

D. INBOARD Containment Building Supply Isolation Bypass Valve, 1VR002B

Answer: A

Explanation:

A is correct – Per CPS 4001.02C001, page 7, this is a Group 8 valve. Per LP85407, page 58, Div 2 NSPS services the Load Driver for the Inboard valves. A ZERO input to the Load Driver produces an energized function to close the Inboard valves.

B and C are incorrect – These Outboard valves would close if the malfunction were to occur in Div 1 NSPS.

D is incorrect – Although an Inboard (Div 2) valve, this valve belongs to Group 9 (see CPS 4001.02C001, page 9). Group 9 is not among the other Groups (10, 12, 16, 19, and 20) that are serviced by the same Load Drivers as Groups 8 and 15.

Objective:	Question Source:	Level of Difficulty:
LP85407.1.7	New	3.0

References provided to examinee:	None
References:	LP85407, Containment and Reactor Vessel Isolation Control System CPS 4001.02C001, Automatic Isolation Checklist

Date Written:	03/01/05	Author:	Ryder

Comments:

At first glance, there <u>appears</u> to be overlap with the RO Simulator JPM portion of the Operating Exam, specifically concerning the following JPM: 40010201LSF01, Verify Group 8 Automatic Isolations. However, there is <u>not</u> overlap, for the following reasons:

1. The relationship between this specific Load Driver and the many valve Groups that it services; i.e., <u>not</u> all of the valves that could actually go closed when the INPUT signal fails to zero are Group 8 valves.

2. That JPM is handed to the Candidate in a condition where the Auto Isolation checklist has already been completed for a number of the Group 8 valves. The Candidate will <u>not</u> have had reason to consider review

Question #	35

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	223002 K3.22	2.5	2.6	Higher
System/Evolution Name:			**Category:**			
PCIS/Nuclear Steam Supply Shutoff System (NSSSS)			Plant Systems			
KA Statement:						
Knowledge of the effect that a loss or malfunction of the PCIS/NSSSS will have on the following: Containment drainage system						

all of the Group 8 valves during that JPM performance.

3. Therefore, *it is very unlikely that the following scenario will occur*…the RO Candidate easily discounts the 'D' distracter because he/she 'remembers' (from the JPM performance) that 1VR002B is not a Group 8 valve.

There is no overlap with the Operating Exam.

Question #	36

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	223001 2.4.6	3.1	4.0	Higher
System/Evolution Name:			**Category:**			
Primary Containment System and Auxiliaries			Plant Systems			
KA Statement:						
Knowledge of symptom based EOP strategies						

Using the provided references, answer the following.

Operators are implementing EOP-1, RPV Control, and EOP-6, Primary Containment Control.

Which ONE of the following conditions REQUIRES operators to START the Hydrogen Mixers, or PERMITS the operators to keep the Hydrogen Mixers running if already started?

A. Igniters are still OFF, THEN hydrogen monitors come on-line after warm-up, and Containment hydrogen reads 9%.

B. Hydrogen Mixers are still OFF, THEN hydrogen monitors come on-line after warm-up, and both Drywell and Containment hydrogen read 2%.

C. Igniters are still OFF, THEN hydrogen monitors come on-line after warm-up, and Containment hydrogen reads 8% with Containment Pressure at 10 psig.

D. Hydrogen Mixers are still OFF, THEN hydrogen monitors come on-line after warm-up, with Drywell hydrogen reading <0.5% and Containment hydrogen declared to be UNKNOWN.

Answer: B

Explanation:
B is correct – Once EOP-7 is entered (in this case, on detectable hydrogen), only EOP-7 can cause them to be started. With both drywell and containment hydrogen reading 2%, operators proceed straight down the left-leg of EOP-7, proceed through the left-most WAIT step and start the Mixers. A is incorrect – This choice puts the Candidate solidly into EOP-7, specifically at the right-most leg of EOP-7, where with the Igniters still OFF, operators are directed to stop the Mixers and prevent igniter restart. C is incorrect – Per EOP-7, Figure R, the plant is above the Deflagration Limit, requiring the implementation of the top-most override step and the execution of the right-most leg, where operators are directed to prevent igniter restart and stop the Mixers. D is incorrect – Once EOP-7 is entered (in this case, because of UNKNOWN Containment hydrogen), only EOP-7 can cause them to be started. This choice is similar to the correct answer choice, 'B', except that the Candidate is still WAITING for detectable Drywell hydrogen (.5% or higher) before the Mixers can be started.

Question #	36

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	2	223001 2.4.6	3.1	4.0	Higher
System/Evolution Name:			**Category:**			
Primary Containment System and Auxiliaries			Plant Systems			
KA Statement:						
Knowledge of symptom based EOP strategies						

Objective:	Question Source:	Level of Difficulty:
None	New	3.6

References provided to examinee:	EOP flowcharts
References:	CPS EOP-6, Primary Containment Control CPS EOP-7, Hydrogen Control

Date Written:	05/02/05	**Author:**	Ryder
Comments: None			

Question #	37

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295001 AK1.01	3.5	3.6	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of Forced Core Flow Circulation	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

With the plant in MODE 4, which ONE of the following describes an operational implication of a COMPLETE loss of Shutdown Cooling, COINCIDENT WITH having NO Reactor Recirc Pumps available?

A. FC will have to be lined up for Alternate Shutdown Cooling.

B. RCIC will have to be lined up for Alternate Shutdown Cooling.

C. RPV water level will have to be maintained ABOVE the steam separators.

D. RPV water level will have to be maintained BELOW the steam separators.

Answer: C

Explanation:

C is correct – Per CPS 4006.01, Section 4.6. Maintaining level above 44" Shutdown Range will provide for some core cooling via natural circulation. Per LP 85422, page 7, this level corresponds to a water level above the steam separators.

A is incorrect – Per CPS 4006.01, Table 2. This lineup requires the RPV head be off (plant in Mode 5).

B is incorrect – Per CPS 4006.01, Table 2. In Mode 4, temperature is 200°F or less, well below the 60 psig RCIC system isolation point.

D is incorrect – See the explanation for the correct answer, 'C'.

Objective:	Question Source:	Level of Difficulty:
None	New	2.7

References provided to examinee:	None
References:	LP85422, Reactor Vessel and Internals CPS 4006.01, Loss of Shutdown Cooling

Date Written:	04/29/05	Author:	Ryder

Comments:

This closed-reference question is categorized as Higher Cognitive (HCL) because:

1. The correct answer is neither an Immediate Operator Action, nor a Precaution/Limitation, where the Candidate would be expected to recall such from memory.
2. The stem requires the Candidate to associate the loss of forced core circulation with the need to ensure adequate natural circulation as a substitute.
3. The elimination of the distracters requires the several 'associations' described in the Explanation for each

Question #	37

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295001 AK1.01	3.5	3.6	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of Forced Core Flow Circulation	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

(above).

Additionally, this question is on the RO Exam (and is not reserved as SRO-only), because the question is not actually asking the Candidate to direct an action found in the Subsequent Actions section of the off-normal (Loss of SDC). Rather, it is framed in such as way that it simply makes use of that Subsequent Action (in Section 4.6) to require the Candidate to demonstrate an understanding of the operational concern that results from having no forced reactor coolant flow.

Question #	38

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295005 AA1.01	3.1	3.3	Higher

System/Evolution Name:	Category:
Main Turbine Generator Trip	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system

The plant is operating at rated power when the main turbine trips (cause unknown).

WITHOUT operator action, which ONE of the following describes the status of main control room indications related to Reactor Recirculation (RR), 20 SECONDS AFTER the reactor scrams?

At P680...

A. the GREEN lights are lit for RR Pump breakers CB5A and CB5B.

B. Total Core Flow indicates about 45 mlbm/hr.

C. both RR FCVs indicate about 10% open.

D. both RR FCV LIMITER ERROR meters read upscale (POSITIVE).

Answer: A

Explanation:

A is correct – Per LP85202, pages 23, 28, 29, and Figure 8. Whenever the turbine trip scram is enabled (above 33.3% power, nominally), the EOC-RPT trip is also enabled. The turbine trip automatically downshifts both RR Pumps to SLOW speed. Operators will see the red (CLOSE) light extinguish and the green (OPEN) light illuminate for each pump's 6.9KV (Fast speed) breaker, CB5A(B).

B is incorrect – Per CPS 9041.01, Figures 1a, 1b, and 2a, this is the expected total core flow that results from a Flow Control Valve Runback to about 19% open indication with both RR Pumps still running in FAST speed. The resulting Total Core Flow indication with both pumps running at SLOW speed (after the EOC-RPT trip), with the FCV still nearly wide open, is only about 29 mlbm/hr (this value obtained from simulator modeling).

C is incorrect – FCVs remain as is during the downshift (typically, indicating about 76% open). Even if the post-scram level/level control transient were to result in a FCV Runback signal being produced, the FCV will indicated about 19% open, not 10% open. The '20 Seconds after' statement in the stem allows for the following: 1) the 5-second delay between the scram signal and a generator reverse power trip (see LP85461, page 33), and 2) enough time thereafter to suggest that the FCVs need some time to stoke to a 10% open position.

D is incorrect – The only time the Limiter Error can read on the Positive side of zero (mid-scale), is when a position indication failure has occurred (e.g., LVDT or RVDT feedback signal is sending false position information to the controller). Given that the FCVs haven't moved in this scenario, the Error will still read zero (mid-scale).

Objective:	Question Source:	Level of Difficulty:
LP85202.1.10.7	New	3.1

Question #	38

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	*Cog Level*
Both	1	1	295005 AA1.01	3.1	3.3	Higher

System/Evolution Name:	**Category:**
Main Turbine Generator Trip	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system

References provided to examinee:	None
References:	LP85202, Reactor Recirculation System LP85461, Main Generator System CPS 9041.01, Jet Pump Operability Test

Date Written:	04/29/05	**Author:**	Ryder
Comments: None			

Question #	39

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295004 2.1.14	2.5	3.3	Lower

System/Evolution Name:	Category:
Partial or Complete Loss of D.C. Power	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of system status criteria which require notification of plant personnel

Per CPS 4201.01, Loss of DC Power, which ONE of the following DC MCC losses requires that the FIRE PROTECTION GROUP be contacted because of the impact on Fire Protection/Detection System equipment requiring that power source?

A. 1A

B. 1B

C. 1C

D. 1D

Answer: B

Explanation:
B is correct - Per CPS 4201.01, Section 4.2.2. DC MCC 1B loss requires notification of Fire Protection Group to assess the impact on fire systems. Per CPS 4202.01C002, Load Impact List, page 5, this bus powers the fire panels. This impact is also described in LP85286, page 95 (DC MCC 1B is Div II power). A, C, and D are incorrect – For the reasons described above. Attached, here, are the Load Impact Lists for these 3 buses, verifying there is no direct cause-effect relationship with fire systems, when either of these 3 buses are lost.

Objective:	Question Source:	Level of Difficulty:
LP85286.1.10.12	New	3.7

References provided to examinee:	None
References:	LP85286, Fire Protection and Detection CPS 4201.01, Loss of DC Power CPS 4201.01C001(2,3,4), Loss of 125VDC MCC 1A(B, C, D) Load Impact Lists

Date Written:	03/02/05	Author:	Ryder

Comments:

Although the SRO would direct the subsequent actions of this Abnormal Operating Procedure (4201.01), including identifying the need to contact Fire Protection for this bus loss, this question is, nonetheless, categorized as BOTH (RO and SRO) for the following reasons:

1. ROs are expected to recognize the <u>major impact</u> of all significant bus losses, especially those <u>vital</u> (Divisional, Class 1E) bus losses such as the DC MCCs 1A-1D, <u>without</u> the need to consider what may or may not be described in a procedure.

2. The <u>uniqueness</u> of the relationship between <u>this</u> bus loss (Division 2) and the fire systems, among these 4 buses, is essentially <u>systems</u> knowledge, and is stated <u>explicitly</u> in the Fire Protection lesson plan, LP85286, and is tied to a Learning Objective (.1.10.12) for which <u>both</u> the RO and SRO are responsible.

Question #	40

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295006 AK2.07	4.0	4.1	Lower

System/Evolution Name:	Category:
SCRAM	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between SCRAM and the following: Reactor pressure control

Following a reactor scram from 40% power, operators are automatically controlling reactor pressure with the turbine bypass valves.

Per CPS 4100.01, Reactor Scram, which ONE of the following identifies the VALUE to where operators should lower reactor pressure in order to minimize the Feedwater-to-RPV Differential Temperature while cooling down?

A. 800 psig

B. 700 psig

· C. 600 psig

D. 500 psig

Answer: C

Explanation:
C is correct – Per CPS 4100.01, Section 4.2 NOTE. Operators should lower reactor pressure to about 600 psig to minimize the RPV-to-FW delta-T that exists during a plant cooldown. A, B, and D are incorrect – For the reason described above.

Objective:	Question Source:	Level of Difficulty:
None	New	2.7

References provided to examinee:	None
References:	CPS 4100.01, Reactor Scram

Date Written:	04/29/05	Author:	Ryder

Comments: None

At first glance, this question appears to have a potential Operating Exam overlap problem. While it is true that one or more Simulator Scenarios may progress to where a comparable pressure control band (550-650 psig) may be *established for* EOP *mitigation strategy* purposes, this is purely coincidental relative to this written exam question. This question is framed entirely in the context of EOP-free, post-scram pressure control strategies, directed solely from the SCRAM abnormal operating procedure (4100.01). **This question does NOT overlap with the Operating Exam.**

Question #	41

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295016 2.1.32	3.4	3.8	Lower
System/Evolution Name:			**Category:**			
Control Room Abandonment			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to explain and apply system limits and precautions						

Following an evacuation of the main control room, operators are establishing control of the plant per CPS 4003.01, Remote Shutdown.

Which ONE of the following describes an associated operator action, and the reason for that action?

A. If placing RCIC in service for level control, RUN the RCIC pump on RECIRC FLOW at about 60 gpm for AT LEAST 2 MINUTES; this ensures the pump is properly warmed up, without damaging pump internals.

B. BEFORE starting SX Pump 1A, CLOSE the PSW To SSW 1A Header Isolation Valve, 1SX014A; this preempts the consequence of a potential 'hot short' condition that might prevent the valve from automatically closing.

C. If using LPCI 'B' for injection, dispatch an operator to DETERMINE PUMP D/P FROM LOCAL INDICATIONS; this is the ONLY way to determine pump flow rate, if degraded flow conditions are suspected.

D. BEFORE attempting to place RHR 'B' in Shutdown Cooling, VERIFY that Shutdown Cooling Outboard Suction Isolation, 1E12-F008, CAN BE OPENED; the valve may be disabled due to a 'hot short' condition.

Answer: C

Explanation:

C is correct – Per CPS 4003.01C011, Section 3.0. There is no RHR Pump 'B' flow indication or motor amps indication at the Remote Shutdown Panel (RSP). This is the only way to determine pump flow rate, should such information be needed. Suspicion of 'degraded flow conditions' would be a need for such information; hence, the 'required' action.

A is incorrect – Per CPS 4003.01C002, Section 3.5, running the pump on Recirc (min flow) should be limited to <20 seconds.

B is incorrect – Refer to CPS 4003.01C005, Section 3.0. There is no such requirement for manually shutting this valve before the pump start; rather, the operator is directed to verify it auto-closes when the pump starts.

D is incorrect – Refer to CPS 4003.01C013, Section 1.4. Operators are cautioned to beware that a hot short could have disabled 1E12-F009 (the Inboard SDC isolation). Although both valves, 1E12-F009 and F008 are potentially vulnerable to hot short problems, notice Section 4.8 of this procedure. Operators must unlock and close (place to ON) the motor breaker for F008. The way that CPS has addressed the 'hot short' problem common to both valves (F008 and F009) is by keeping the F008 breaker locked open during normal plant operating conditions (see CPS 3312.03, Section 6.3). Hence, the Remote Shutdown Div 2 SDC procedure (4301.01C013) is silent on the need for operators to be concerned about the status of F008. The valve is administratively protected from 'hot short' vulnerability.

Question #	41

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295016 2.1.32	3.4	3.8	Lower

System/Evolution Name:	Category:
Control Room Abandonment	Emergency and Abnormal Plant Evolutions

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

Objective:	Question Source:	Level of Difficulty:
None	New	3.2

References provided to examinee:	None
References:	CPS 4003.01, Remote Shutdown CPS 4003.01C002, RSP – RCIC Operation CPS 4003.01C005, RSP – Div 1 SX Operation CPS 4003.01C011, RSP – Div 2 LPCI Operation CPS 4003.01C013, RSP – Div 2 Shutdown Cooling Operation

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	42

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295019 AK2.11	2.5	2.6	Higher
System/Evolution Name:			**Category:**			
Partial or Complete Loss of Instrument Air			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Radwaste						

The plant is operating at RATED POWER when the following occurs:

- An air line rupture occurs in the RADWASTE BUILDING Instrument Air Ring Header
- BOTH of the Radwaste Bldg IA Header Isolation Valves, AND BOTH of the Radwaste Building SA Header Isolation Valves SHUT
- NO other building Service Air or Instrument Air supplies are affected
- ALL other air header isolation valves are STILL OPEN

Which ONE of the following describes the plant/system impact of this loss of air?

A. FC Demineralizers isolate.

B. Main condenser vacuum slowly degrades.

C. WO Chillers automatically shut down.

D. ACTUAL D/P's on the CW Traveling Screens RISE.

Answer: A

Explanation:

A is correct – Per CPS 5041-4B, the Fuel Pool Cleanup (FC) demins isolate.

B is incorrect – Per CPS 5041-4B. Because of the failure mode of the 1N66-F060 valve (fails 'as is') Off-gas is unaffected, and therefore condenser vacuum is unaffected. This choice is attractive to the Candidate who does recall that the Mechanical Vacuum Pump (MVP) Separate Tank Vent Valve will SHUT on this loss of air, causing a blow out of loop seal, and a loss of vacuum, IF the MVP were in service. With the plant at rated power, the MVPs are not in service, and they are isolated.

C is incorrect –Per CPS 5041-4C, these chillers are loads on the Control Building IA Ring Header. With stem conditions indicating there was a successful isolation of the RW Bldg air headers, leaving all others unaffected, these chillers should be unaffected.

D is incorrect – Per CPS 5041-5E, the traveling screen air bubblers are loads on the Turbine Building SA Ring Header. For the same reason as discussed in choice 'C', these bubblers should continue to function as designed. The choice is distracting in that is describes how ACTUAL screen d/p would trend if air were lost to the bubblers…i.e., screens would not auto-start (or auto-shift to Fast speed) on high d/p; consequently, ACTUAL screen d/p will rise.

See Figure 1 of LP85301 for a simplified view of how these air headers are arranged.

Objective:	Question Source:	Level of Difficulty:
LP85301.1.11.3	New	2.6

Question #	42

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295019 AK2.11	2.5	2.6	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of Instrument Air	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Radwaste

References provided to examinee:	None
References:	LP85301, Service and Instrument Air CPS 5041-4B, 4C, and 5E, alarm response procedures

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	43

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295023 AK2.02	2.9	3.2	Higher
System/Evolution Name:			**Category:**			
Refueling Accidents			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Fuel pool cooling and cleanup system						

Operators have JUST BEGUN transferring 20 spent fuel bundles from the Containment Transfer Pool to the Spent Fuel Storage Pool, when the only available FC Pump trips and CANNOT be restarted.

Which ONE of the following describes a reason why these spent fuel transfers must be STOPPED until an FC Pump can be placed back in service?

A. Area radiation levels on CNMT el. 828' will rise with EACH fuel bundle transferred.

B. A FULL carriage in the IFTS transfer tube will raise temperature RAPIDLY enough to damage the fuel before it exits.

C. Spent Fuel Storage Pool temperature will rise RAPIDLY with EACH fuel bundle placed in the pool.

D. Area radiation levels in accessible areas around the IFTS will rise to potentially LETHAL levels.

Answer: A

Explanation:

A is correct – Per LP85233, pages 22, 39, and 46. Without forced FC flow, there is no way to replenish the 1,000 gallons (approximate) removed from the Containment Transfer Pool (on CNMT el. 828') each time the IFTS transfer tube is flooded for carrying a fuel carriage down to the Fuel Building Transfer Pool. As Containment Transfer Pool level lowers, with each fuel bundle(s) transferred, area radiation levels will rise, creating a local radiological hazard for personnel on CNMT el. 828'.

B is incorrect – Per CPS 3702.01, Section 2.1.5, the transfer tube is designed to handle fuel for up to 10 hours without additional cooling.

C is incorrect – Per CPS 3317.01, Section 4.14, even at high heat loads in the spent fuel storage pool, degraded FC conditions will result in pool temperature rising on the order of 'several' degrees per hour.

D is incorrect – Per CPS 3702.01, Section 2.1.7, lethal radiation levels exist in these spaces <u>whenever</u> fuel is transiting the transfer tube, regardless of the status of FC.

Objective:	Question Source:	Level of Difficulty:
LP85233.1.13.2	New	3.3

Question #	43

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295023 AK2.02	2.9	3.2	Higher

System/Evolution Name:	Category:
Refueling Accidents	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Fuel pool cooling and cleanup system

References provided to examinee:	None
References:	LP85233, Fuel Pool Cooling and Cleanup System CPS 3702.01, Inclined Fuel Transfer System (IFTS) CPS 3317.01, Fuel Pool Cooling and Cleanup System

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	44

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295025 EK3.09	3.7	3.7	Lower

System/Evolution Name:	Category:
High Reactor Pressure	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the reasons for the following responses as they relate to HIGH REACTOR PRESSURE: Low-low set initiation

Which ONE of the following identifies a design feature that acts to prolong the life of MOST of the 16 Safety Relief Valves?

A. Overpressure Relief mode of SRV operation

B. Low-Low Set initiation

C. Overpressure Safety mode of SRV operation

D. ADS initiation

Answer: B

Explanation:
B is correct – Per LP85239, page 16. The 5 LLS SRVs act to reduce the number of SRVs that cycle for given plant conditions, prolonging SRV life. A, C, and D are incorrect – See LP85239, pages 11-13. Neither of these functions are directly related to prolonging SRV life.

Objective:	Question Source:	Level of Difficulty:
None	New	2.7

References provided to examinee:	None
References:	LP85239, Main Steam System

Date Written:	03/04/05	Author:	Ryder
Comments: None			

Question #	45

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295026 EK3.02	3.9	4.0	Higher

System/Evolution Name:	Category:
Suppression Pool High Water Temperature	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the reasons for the following responses as they relate to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Which ONE of the following identifies an ADVANTAGE of taking the EOP-6 action to 'Start all available pool cooling' when Suppression Pool Temperature is 97°F and slowly rising?

A. Extends the time that it remains acceptable to INITIATE Containment Sprays.

B. Extends the time before HAVING to inject boron, if shutdown criteria is NOT met, but reactor power is BELOW 5%.

C. ENSURES that RCIC Pump damage due to inadequate NPSH will NOT occur if the pump is taking a suction from the suppression pool.

D. ENSURES that the Containment design temperature limit will NOT be exceeded while the rate of blowdown energy transfer is greater than the containment venting capacity.

Answer: B

Explanation:
B is correct – Refer to EOP-1A, Power leg. Any attempt to slow down the rate of suppression pool temperature rise will delay the <u>requirement</u> to start SLC <u>before</u> reaching the Boron Injection Temperature (BIT) of Figure G. A is incorrect – Refer to Figure O of EOP-6. This choice provides face validity in light of the unavailability of a 3rd distracter that points directly at suppression pool temperature. It provides sufficient distraction, in that it readily attracts one to a very familiar EOP Figure, familiar even to the weakest of Candidates; where the other 3 choices demand a greater investment of time and analysis to determine their specific relationship with the given pool temperature. C is incorrect – Refer to EOP-1, Figure Z. Even if operators are able to STOP the rise in pool temperature (by starting all available pool cooling, alone), and therefore stay far below 197°F, it does <u>not</u> ENSURE that RCIC pump cavitation is avoidable. Figure Z clearly shows that either too low a pool level, or too high a pump flow, can <u>still</u> lead to cavitation and pump damage. D is incorrect – Refer to EOP-6, Figure P, and to EOP Technical Bases, Section 12-H. This choice suggests that the Heat Capacity Limit is dependent solely on suppression pool temperature. It is not. Even if '*starting all available pool cooling*' were able to allow pool temperature to rise no higher than, for example, 140°F, a low enough pool level (15.1 feet, in this case) would still exceed the HCL, which is defined by this choice (a paraphrase of Section 12-H of the Bases).

Objective:	Question Source:	Level of Difficulty:
LP87553.1.6.8	New	3.7

Question #	45

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295026 EK3.02	3.9	4.0	Higher
System/Evolution Name:			**Category:**			
Suppression Pool High Water Temperature			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of the reasons for the following responses as they relate to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling						

References provided to examinee:	EOP flowcharts
References:	CPS EOP-1, RPV Control CPS EOP-1A, ATWS RPV Control CPS EOP-6, Primary Containment Control CPS EOP Technical Bases

Date Written:	03/04/05	**Author:**	Ryder

Comments:

This question is categorized as Higher Cognitive (HCL) for the following reasons:

1. The correct answer demands that the Candidate associate the given Suppression Pool Temperature (97°F and rising) with the Boron Injection Initiation Temperature (BIT) derived from Figure G of EOP-1A. The association must recognize that with only 97°F pool temperature, no matter what the power level, there is still time to potentially delay boron injection by slowing down the rate of pool temperature rise.

2. The elimination of, at least, the choice 'D' distracter demands that the Candidate recognize this choice is describing the operational meaning of the Heat Capacity Limit, and then recognize that a 97°F pool temperature is well below even the most limiting portions of Figure P.

Question #	46

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295027 EA2.01	3.7	3.7	Higher
System/Evolution Name:			Category:			
High Containment Temperature			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE: Containment Temperature						

Using the provided references, answer the following.

A Station Blackout (SBO) is in progress, with the following:

- Suppression Pool Level reads 18 feet on its ATM
- Containment Pressure reads 2.5 psig and slowly rising on its ATM

Per CPS 4200.01, Loss of AC Power, ONLY ONE of the following identifies information, about the CURRENT Containment Temperature, that operators should expect to have AVAILABLE and be ABLE to use:

1. Containment Temperature reads 165°F and slowly rising on its ATM.
2. Inside Containment, a hand-held barometer reads 1.5 psia and slowly rising.
3. Inside Containment, a hand-held infrared thermometer reads 122°F and slowly rising.
4. Containment Temperature reads 190°F and slowly rising on a portable resistance-temperature bridge.

Which ONE of the following describes where operators CURRENTLY are in EOP-6, with respect to the CONTAINMENT TEMPERATURE leg?

A. Monitoring for a possible EOP-6 entry on Containment Temperature

B. At the BOTTOM-most IF-THEN step, about to proceed to EOP-3

C. At the TOP-most IF-THEN step, waiting for some AC power restoration

D. Have just determined it would be OK To Spray, if an RHR Pump were available

Answer: B

Explanation:
B is correct – Per CPS 4200.01C003, page 9. I&C personnel use a portable bridge connected to the permanent RTDs at designated cabinet terminal points, extrapolate the corresponding containment temperature and report it to control room operators. With 190°F and slowly rising, operators are at the bottom-most IF-THEN step of the Containment Temperature leg.

Question #	46

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295027 EA2.01	3.7	3.7	Higher

System/Evolution Name:	Category:
High Containment Temperature	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE: Containment Temperature

A is incorrect – This would be the choice for a Candidate who believes: 1) the prescribed method employs sending personnel into a hot containment with a barometer, and 2) the atmosphere is a saturated one, allowing operators to extrapolate containment temperature using the Steam Tables. The alleged corresponding containment temperature would be 116°F. If this were true, operators would continue to monitor containment temperature for a possible EOP-6 entry at 122°F.

C is incorrect – This would be the choice for a Candidate who believes the prescribed method employs sending personnel into a hot containment with an infrared thermometer. It is <u>not</u> the method prescribed by the procedure. If it were, with 122°F and slowly rising, operators would be at the top-most IF-THEN step of the Containment Temperature leg.

D is incorrect – This would be the choice for a Candidate who believes there is an ATM instrument for Containment Temperature; per CPS 4200.01, Appendix B, there is <u>not</u>. If there were, at 165°F and slowly rising, with Containment Pressure at 2.5 psig, Figure O would indicate that it is OK to Spray; operators would be waiting for an available RHR Pump.

Objective:	Question Source:	Level of Difficulty:
None	New	3.3

References provided to examinee:	EOP flowcharts Steam Tables, pages 5-6, photo-copy, enlarged
References:	Steam Tables CPS 4200.01C003, Monitoring CNMT Temperatures During a SBO EOP-6, Primary Containment Control

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	47

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295028 2.4.4	4.0	4.3	Lower
System/Evolution Name:				**Category:**		
High Drywell Temperature				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures						

Which ONE of the following events requires an entry into one or more of the EOP FLOWCHARTS?

A. After a scram, the Shutdown Criteria are NOT met.

B. After a scram, reactor water level shrinks to +15 inches before recovering.

C. The VF Exhaust CAM, 1RIX-PR019, reaches its ALERT alarm point.

D. A break in an SRV discharge connection raises Drywell temperature to 155°F.

Answer: D

Explanation:

D is correct – Per EOP-6. Drywell temperature entry is required at 150°F.

A is incorrect – Per EOP-1. Unless there is first an EOP-1 entry (there may not be on a low-power scram), there is no way into EOP-1A for failure to meet shutdown criteria. Rather, the REACTOR SCRAM abnormal, CPS 4100.01, Section 4.3.1, directs operators to use one of the EOP support procedures, CPS 4411.08, to alternatively insert control rods.

B is incorrect – Per EOP-1. Unless level drops to Level 3 (+8.9 inches), no EOP entry is required. A low-power scram may not shrink level that low.

C is incorrect – Per EOP-8. This CAM must reach its HIGH alarm point before an EOP entry is required.

Objective:	Question Source:	Level of Difficulty:
LP87558.1.1	New	2.6

References provided to examinee:	*EOP flowcharts*
References:	EOP-1, RPV Control EOP-1A, ATWS RPV Control EOP-6, Primary Containment Control EOP-8, Secondary Containment Control CPS 4100.01, Reactor Scram

Date Written:	03/07/05	Author:	Ryder

Comments:

Although plant events on the Simulator Scenario portion of the Operating Exam will require operators to enter EOP-6, at no time will the <u>initiating</u> event be one that requires operators to recognize a 150°F drywell temperature as, uniquely, the reason for entry. As such, this written exam question does <u>not</u> overlap with the Operating Exam.

Question #	48

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	2950131 EK1.03	3.7	4.1	Higher

System/Evolution Name:	Category:
Reactor Low Water Level	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power

Consider EOP-1A, ATWS RPV Control, when answering the following.

WITH reactor power still ABOVE 5%, which ONE of the following identifies an INDICATED reactor water level that can be characterized by the following statement?

> "Core Void Fraction is relatively LOW; nonetheless, FUEL DAMAGING power oscillations are NOT EXPECTED."

A. – 40 inches

B. – 65 inches

C. – 140 inches

D. – 162 inches

Answer: B

Explanation:
B is correct – Per EOP Technical Bases, EOP-1A, pages 5-14, 5-15, 5-17, and 5-18. This level (-65") is just below the level at which the feedwater sparger is fully uncovered, thus reducing the core inlet subcooling by 65-75%, a point where large-scale core instabilities are <u>not</u> expected to occur. However at this same level (-65"), Void Fraction inside the core shroud is still 'relatively' low. This is due to there still being a sufficient head of water that sustains a good amount of natural circulation (note: RR Pumps tripped at –45"). At –65", this natural circulation continues to sweep voids up and out, resulting in a 'still relatively low Void Fraction. Per pages 5-17 and 5-18, only when operators continue to lower level, will the natural circulation contribution be reduced to a point where the core voids outs (i.e., a HIGH Void Fraction will exist inside the core shroud). A is incorrect – For the reasons described above. At this level (-40") RR Pumps are still running in SLOW speed (see EOP Technical Bases, page 5-16). Regardless of comparing void contents, fuel damaging core instabilities (power oscillations) at high power and low flow conditions is <u>still</u> a major concern because there remains a good amount of core inlet subcooling with level this high above the feedwater sparger. C and D are incorrect – For the reasons associated with the correct answer. At these low levels (-140" and –162"), fuel damaging power oscillations are NOT expected. However, the whole purpose of intentionally controlling level this low (in either Band 'B' or Band 'C') is to <u>increase</u> the void fraction <u>inside</u> the shroud, thus keeping power down. Again, refer to pages 5-17 and 5-18.

Objective:	Question Source:	Level of Difficulty:
LP87553.1.3	New	3.5

References provided to examinee:	None required, but access to EOP flowcharts is OK
References:	CPS EOP-1A, ATWS RPV Control CPS EOP Technical Bases

Question #	48

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	2950131 EK1.03	3.7	4.1	Higher

System/Evolution Name:	Category:
Reactor Low Water Level	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	49

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295022 AA1.04	2.5	2.6	Higher
System/Evolution Name:			Category:			
Loss of CRD Pumps			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: Reactor water cleanup system						

The plant is in MODE 4, with the following:

- Operators are maintaining a STEADY reactor water level with the following:
  - o CRD Pump 'A' is feeding the RPV
  - o RWCU is rejecting to the main condenser at 45 gpm, using ONLY the 'low flow' valve
- CRD Pump 'B' is UNAVAILABLE
- THEN, CRD Pump 'A' trips and CANNOT be restarted

Per CPS 3303.01, Reactor Water Cleanup, which ONE of the following describes the operator action with respect to RWCU?

A. CLOSE 1G33-F041, Drain Flow to Condenser Bypass; THEN, CLOSE 1G33-F031, Drain Flow Orifice Bypass, and FULLY CLOSE 1G33-F033, Drain Flow Regulator.

B. CLOSE 1G33-F046, Drain Flow to Condenser; THEN, VERIFY CLOSED 1G33-F031, Drain Flow Orifice Bypass, and FULLY CLOSE 1G33-F033, Drain Flow Regulator.

C. AS NECESSARY to maintain the desired reactor water level, THROTTLE closed 1G33-F033, Drain Flow Regulator.

D. AS NECESSARY to maintain the desired reactor water level, THROTTLE closed 1G33-F031, Drain Flow Orifice Bypass.

Answer: A

Explanation:

A is correct – Stem conditions indicate that reject is lined up per CPS 3303.01, Section 8.1.6.3.1, for reactor pressure <50 psig (Mode 4), with the 'low drain flow' valve, F041, open, the high drain flow valve, F046, still shut, and the Drain Flow Orifice Bypass, F031, fully open. With a 'steady' reactor water level being maintained, a trip of the CRD pump would require that the reject flow path be fully secured to maintain level. Therefore, operators implement Section 8.1.6.9. In this case, they will close F041, then close F031, and fully close F033.

B is incorrect – For the reasons described above. 1G33-F046 was <u>not</u> open for this <u>low</u> reject flow lineup.

C is incorrect – Candidate is expected to recognize the pre-trip flow balance that existed (45 gpm in, 45 gpm out). As such, any attempt to throttle down on 1G33-F033 in an effort to maintain water level would take reject flow below 13 gpm. Per CPS 3303.01, Section 8.1.6.3 CAUTION (page 37), this may result in a system isolation on delta-flow.

Question #	49

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295022 AA1.04	2.5	2.6	Higher

System/Evolution Name:	Category:
Loss of CRD Pumps	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: Reactor water cleanup system

D is incorrect – Only if the plant were above 50 psig, would operators possibly have the F031 valve in a throttled open position, before the pump trip. Even if it were to be used, attempting to maintain level with this valve would be incorrect for the same reason as described for choice 'C'.

Objective:	Question Source:	Level of Difficulty:
LP85204.1.3.2	New	3.7

References provided to examinee:	None
References:	CPS 3303.01, Reactor Water Cleanup System CPS 3304.01, Control Rod Hydraulic & Control (RD)

Date Written:	05/02/05	Author:	Ryder
Comments: None			

Question #	50

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295029 2.1.33	3.4	4.0	Lower

System/Evolution Name:	Category:
High Suppression Pool Water Level	Emergency and Abnormal Plant Evolutions

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

The plant is in MODE 3 with reactor pressure at 140 psig.

Which ONE of the following events requires a Technical Specification ENTRY?

A. From a standby lineup, the RCIC Turbine Trip Throttle Valve, C002E, trips SHUT due to a BROKEN latch-trip hook assembly.

B. TWO of the level transmitters for the Scram Discharge Volume High Level Trip Function FAIL their Channel Calibration.

C. While terminating Suppression Pool makeup, the Supp Pool Fill Valve, 1SM004, sticks open; pool level rises to 20 FEET, 5 INCHES, before operators can stop the rise.

D. An electrical short and fire DESTROYS the MOTOR-OPERATOR for RHR B Shutdown Cooling Suction Valve, 1E12-F006B; the fire is quickly extinguished.

Answer: C

Explanation:
C is correct – Per Tech Spec 3.6.2.2. Mode 3, below 235 psig, upper level limit is 20 feet, 1 inch. A is incorrect – Per Tech Spec 3.5.3. In Mode 3, RCIC operability is not required until reactor pressure is above 150 psig. B is incorrect – Per Tech Spec Table 3.3.1.1-1, Function #8. SDV level transmitter trip function is required <u>only</u> in Modes 1, 2, and 5(a). D is incorrect – Per Tech Spec 3.4.9. In Mode 3, Shutdown Cooling operability is not required unless reactor pressure is below the RHR cut-in permissive pressure. That pressure setpoint is 104 psig (see LP85205, page 20).

Objective:	Question Source:	Level of Difficulty:
LP85223.1.16	New	3.3

References provided to examinee:	None
References:	LP85205, Residual Heat Removal System CPS Tech Spec 3.6.2.2, Suppression Pool Water Level CPS Tech Spec 3.5.3, RCIC CPS Tech Spec 3.3.1.1, RPS Instrumentation CPS Tech Spec 3.4.9, RHR Shutdown Cooling – Hot Shutdown

Date Written:	03/08/05	Author:	Ryder
Comments: None			

Question #	51

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	500000 EK2.01	3.1	3.5	Higher

System/Evolution Name:	Category:
High Containment Hydrogen Concentration	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between HIGH CONTAINMENT HYDROGEN CONCENTRATION and the following: *Containment hydrogen monitoring systems*

Using the provided references, answer the following.

Following a scram on high drywell pressure, operators are placing the Containment H2/O2 monitors in service.

Which ONE of the following identifies the EARLIEST time when operators are PERMITTED TO USE the monitors to determine Containment hydrogen concentration?

20 minutes after...

A. placing the OVEN TEMP SELECT switch in HIGH.

B. observing the OVEN TEMP ABNORMAL light extinguish.

C. the STARTUP CYCLE clock begins counting up.

D. RE-OPENING the associated containment isolation valves.

Answer: C

Explanation:
C is correct – Refer to CPS 4411.11, Section 2.1.5. In order to utilize monitor readings, there must have been 20-minutes run time (i.e., sampling time) with the containment isolation valves open. Step 2.1.1 has operators place the Oven Temp Select switch in HIGH (at which time, the Oven Temp Abnormal light illuminates, signifying that a warm-up has begun). Step 2.1.2 has operators re-open the containment isolation valves (closed on the high drywell pressure scram signal), after about 20-minutes (i.e., when the Oven Temp Abnormal light extinguishes). At this point, the monitors are still not sampling. Only when operators depress the 'Enter' key in step 2.1.4.8 does the sampling time (run time) begin, coincident with the fact that the 270-second 'Startup Cycle' clock begins to count up. A, B, and D are incorrect – *For the reasons described above.*

Objective:	Question Source:	Level of Difficulty:
LP85406.1.9	New	3.0

References provided to examinee:	CPS 4411.11, Hydrogen Control System Operation, **with** the NOTE at the top of page 3 (Section 2.1) **redacted**
References:	CPS 4411.11, Hydrogen Control System Operation

Question #	51

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	500000 EK2.01	3.1	3.5	Higher

System/Evolution Name:	**Category:**
High Containment Hydrogen Concentration	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between HIGH CONTAINMENT HYDROGEN CONCENTRATION and the following: Containment hydrogen monitoring systems

Date Written:	05/16/05	**Author:**	Ryder

Comments:

This open-reference question is not a 'Direct Lookup' type, for the following reasons:

1. The Candidate must first determine which procedure Section to begin with. He/she must recognize (from the stem) that the associated Containment Isolation Valves have gone shut on the high drywell pressure signal; therefore, Section 2.1.2 is where placing the monitors in service begins. To accommodate this first piece, the NOTE at the top of page 3 of the procedure has been **redacted**.

2. The Candidate must recognize (Systems knowledge) the significance of step 2.1.4.8 (as already described above). The fact that this action begins the sampling (run) time, is not explicit in the procedure.

For these same reasons, this question is also categorized as Higher Cognitive.

Question #	52

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295008 AK1.01	3.0	3.2	Higher
System/Evolution Name:				**Category:**		
High Reactor Water Level				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Moisture carryover						

The plant is operating at rated power when a Master Level Controller failure raises reactor water level AS HIGH AS +60 INCHES, and the Level 8 trip function fails.

Which ONE of the following describes the POTENTIAL consequence that is AVOIDED by inserting a manual scram?

A. NON-conservative core thermal power calculation

B. Failure of SRVs to FULLY RE-SEAT after opening

C. Main Turbine excessive vibrations

D. Failure of MSIVs to FULLY CLOSE when needed

Answer: C

Explanation:

C is correct – Refer to LP85422, Figure 21. Instrument Zero elevation is 787 feet, 6 inches. The elevation that corresponds to +60 inches (i.e., +5 feet) is 792 feet, 6 inches. The main steam line nozzle elevation is 797 feet; this is 4 feet, 6 inches above where RPV level rose before operators inserted the scram. The only potential consequence of level this high is a reduced drying efficiency of the moisture separator, and therefore, an increased amount of moisture carryover that could result in high vibration of the main turbine due to water-blade impingement.

A is incorrect - The notion of there being an effect on calculated core thermal power comes from the OPEX discussion in LP85422, page 41. However, as shown in that discussion, a non-conservative situation <u>may</u> be associated with a <u>decreased</u> amount of moisture carryover, <u>not</u> an increased amount.

B is incorrect – With water level getting no where near the main steam lines, there is no chance for affecting the re-seating ability of an SRV.

D is incorrect – With water level getting no where near the main steam lines, there is no chance for affecting the ability of the MSIVs to fully seat on their shut seat due to either water in the steam lines, or unanalyzed D/P's.

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

References provided to examinee:	None
References:	LP85422, Reactor Vessel & Internals

Date Written:	03/11/05	Author:	Ryder
Comments: None			

Question #	53

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295002 AK2.02	3.1	3.2	Higher
System/Evolution Name:			**Category:**			
Loss of Main Condenser Vacuum			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Main turbine						

The plant is operating at 45% power, on the 50% FCL, with the following:

- Circulating Water (CW) Pump 'B' is tagged out for repairs
- THEN, operators perform an emergency shutdown of CW Pump 'C' due to a major oil leak
- CW Pump 'A' remains running
- RO-A lowers reactor power until main condenser vacuum stabilizes
- Main condenser vacuum is now 25" Hg and STEADY
- Reactor power is now 25% and STEADY

Which ONE of the following describes the NEW operational concern that follows this transient?

A. The impact of a potential loss of 6.9 KV Bus 'A'

B. Windage heating of the LP turbine last stage buckets

C. Inability of CW Pump 'C' to automatically trip on high condenser pit level

D. The prompt exit from the Power/Flow Map CONTROLLED ENTRY REGION

Answer: B

Explanation:
B is correct – Post-transient conditions show the plant operating at something less than 300 MWe generator load with too low a condenser vacuum (<26" Hg). Per CPS 4004.02, Section 4.1.1, operating with these conditions should be avoided. Per PB400402, this light load-low vacuum condition causes overheating (due to windage) and moisture erosion of the LP turbine last stage buckets. Applicable calculation for post-transient generator load is as follows: 25% actual / 92.5% max allowed = 'x' MWe / 1052 MWe net 'x' MWe = (.25/.925) X (1052) = 284 MWe A is incorrect – This is no more a concern, now, than it was before the manual trip of CW Pump 'C'. The 6.9 KV 'A' Bus powers both the A and C pumps; a single bus loss with only two pumps running would require a scram. This choice suggests that that vulnerability is greater now (i.e., an IMMEDIATE operational concern) since the 'C' Pump is the only one remaining. This is not true. C is incorrect – Per LP85275, page 28, the high pit level trip circuitry for all 3 CW Pumps is electrically powered by the CW Pump 'A' DC Control power. Because CW Pump is still running, this high pit level capability still exists. D is incorrect – Pre-transient stem conditions (45% power, 50% rod line) are meant cause the Candidate to ponder the effect of having to reduce recirc flow (and/or rod insertions) in order to bring power down to the post-transient 25%. Even without the P/F Map as open-reference, the Candidate is expected to realize that the plant operating well clear of the Controlled Entry Region (the 'prompt' exiting of which would be the IMMEDIATE operational concern).

Question #	53

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295002 AK2.02	3.1	3.2	Higher

System/Evolution Name:	Category:
Loss of Main Condenser Vacuum	Emergency and Abnormal Plant Evolutions
KA Statement:	
Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Main turbine	

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

References provided to examinee:	None
References:	PB400402, Loss of Vacuum LP85275, Circulating Water System CPS 4004.02, Loss of Vacuum CPS 3005.01, Unit Power Changes (for the P/F Operating Map)

Date Written:	05/16/05	**Author:**	Ryder
Comments: None			

Question #	54

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.2.2	4.0	3.5	Lower

System/Evolution Name:	Category:
	Equipment Control

KA Statement:
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels

Per OP-CL-108-101-1001, General Equipment Operating Requirements, which ONE of the following describes how the operator FULLY CLOSES a THROTTLABLE MOV from the main control room?

A. Holds the control switch in the CLOSED position for 1 to 2 seconds after seeing the GREEN light ON with the RED light OFF.

B. After seeing a closed indication, holds the control switch in the OPEN position until an INTERMEDIATE indication is observed, then re-closes the valve.

C. Holds the control switch in the CLOSED position and RELEASES the control switch AS SOON AS the RED light extinguishes.

D. After seeing a closed indication, holds the control switch in the OPEN position for 2 TO 3 SECONDS, then re-closes the valve.

Answer: A

Explanation:
A is correct – Per OP-CL-108-101-1001, Section 3.5.1. This ensures the valve is fully closed. B, C, and D are incorrect – Not the methods described either in <u>this</u> procedure <u>or</u> in <u>any</u> other operating procedure.

Objective:	Question Source:	Level of Difficulty:
None	New	2.0

References provided to examinee:	None
References:	OP-CL-108-101-1001, General Equipment Operating Requirements

Date Written:	03/14/05	Author:	Ryder
Comments: None			

Question #	55

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.3.9	2.5	3.4	Lower

System/Evolution Name:	Category:
	Radiological Controls

KA Statement:
Knowledge of the process for performing a containment purge

The plant is operating at RATED POWER when some PAINTING is to be performed inside the Containment Building.

Which ONE of the following identifies the PREFERRED mode of containment purge operations to support this work?

A. CCP FILTERED Mode

B. Containment Purge Mode

C. Containment Vent Mode

D. CCP UNFILTERED Mode

Answer: D

Explanation:

D is correct – Per CPS 3408.01, Sections 2.2.1 and 4.1. This is the normal operating containment purge mode for Mode 1, and should be used when painting is in progress inside Containment.

A is incorrect – Per CPS 3408.01, Sections 2.1 and 2.1.1. Although this mode is available in Mode 1, it uses the Drywell Purge Filter Trains (DWPFTs), and should not be used because of the concern described in Section 4.1 (i.e., reduced capacity of the charcoal filters due to these chemicals/paint fumes).

B is incorrect – Per CPS 3408.01, Section 2.2.2, this mode uses the Drywell Purge Filter Trains (DWPFTs), and should not be used because of the concern described in Section 4.1 (i.e., reduced capacity of the charcoal filters due to these chemicals/paint fumes). Additionally, this mode is only available in Modes 4 and 5 (see Section 8.1.1.4).

C is incorrect – Although specified by CPS 3408.01, Section 2.2.1 as the preferred mode to be used when painting is being done inside Containment, this mode is only available in Modes 4 and 5 (see Section 8.1.1.3).

Objective:	Question Source:	Level of Difficulty:
LP85455.1.14	New	2.5

References provided to examinee:	None
References:	CPS 3408.01, Containment Building/Drywell HVAC

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	56

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.1.25	2.8	3.1	Higher
System/Evolution Name:			**Category:**			
			Conduct of Operations			
KA Statement:						
Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data						

Using the provided references, answer the following.

ASSUME the following when answering this question:

- Main Power Transformer MVA is EQUAL TO Main Generator MVA

The plant is operating at power, with degraded grid conditions, and the following:

- Main Generator terminal voltage is 20,020 Volts
- Main Generator is operating at a 0.9 power factor with POSITIVE (+) VARs
- 'C' Phase Main Power Transformer (MPT 'C') is operating with NO operable cooling banks
- Outdoor air temperature is 95°F
- MPT 'C' oil and winding temperatures are slowly rising
- Operators have just reduced Main Generator real load (MWe) to lower the MVA load to the MAXIMUM PERMISSIBLE MVAs for these degraded conditions

Which ONE of the following identifies the MAXIMUM amount of time that the Main Generator is permitted to operate with this MVA load?

A. 30 minutes

B. 36 minutes

C. 1 hour, 10 minutes

D. 1 hour, 13 minutes

Answer: A

Explanation:
A is correct – Per CPS 3105.05, Section 8.5.2. 'VARs to the grid' translates to a lagging power factor. At 20,020 volts, the generator is operating at 91% of rated (.91 x 22,000 = 20,020). Section 6.1 Table shows rated voltage is 22,000. Operators determine that the maximum allowable MVA load is 91% of rated MVA. Section 6.1 Table shows Rated MVA is 1179 MVA. Therefore, the maximum allowable MVAs is 1073 MVA (.91 x 1179 = 1073). Per CPS 3504.01, Section 8.3.1, operators recognize that the degraded cooling condition on MPT 'C' is the limiting concern. Operators determine that Table 1 applies (given that no operable cooling banks are available). Per the notes of Section 8.3.1, at 95°F air temperature, operators default to the higher temperature value (104°F), and the higher MVA load value (1140 MVA). *The resulting time limit at this MVA load is 30 minutes.*

Question #	56

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.1.25	2.8	3.1	Higher

System/Evolution Name:	Category:
	Conduct of Operations

KA Statement:
Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data

B is incorrect – For the reasons described above. This choice is plausible if the Candidate inappropriately interpolates the 95F air temperature for the correct Max MVA load (1140 MVA). 95°F is half the interval between 86°F and 104°F; 36 minutes is half the interval between 30 minutes and 42 minutes.

C and D are incorrect – For the reasons associated with the correct answer. These choices are analogous to choice 'A' and 'B', respectively. They are plausible to the Candidate who incorrectly translates the 'VARs *to the grid*' *stem* condition to a leading power factor. Per CPS 3105.05, Section 8.5.2, at 91% of terminal voltage, the MVA load is limited to 83% of rated MVAs, or 978 MVAs (.91 x .91 x 1179 = 978). This Candidate would then apply the '997.5' MVA limit of CPS 3504.01, Section 8.3.1, Table 1.

Objective:	Question Source:	Level of Difficulty:
None	New	3.5

References provided to examinee:	CPS 3105.05, Generator (TG), entire procedure CPS 3504.01, Main Power and UAT Cooling, entire procedure
References:	CPS 3105.05, Generator (TG) CPS 3504.01, Main Power and UAT Cooling

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	57

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295003 AA2.02	4.2	4.3	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of A.C. Power	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Reactor power, pressure, and level

The plant has just scrammed from rated power, THEN the following occurs:

- ALL power is lost from 4160V Bus 1A1
- Operators are controlling reactor pressure between 800 and 1065 psig

Which ONE of the following identifies main control room REACTOR PRESSURE recorders that are available to determine the CURRENT reactor pressure?

A. ONE recorder on P601

B. TWO recorders on P601

C. ONE recorder on P601 AND ONE recorder on P870

D. TWO recorders on P601 AND ONE recorder on P870

Answer: A

Explanation:
A is correct – Per CPS 4200.01, Appendix B, page 24. With 4160 V Bus 1B1 still powered, there is a single reactor pressure instrument still functional (for the current reactor pressure band, 800-1065 psig), at P601. That instrument is B21-R623B (a paperless recorder). The other recorder instrument, 1LR-SM016, shown in this Appendix is a low-range pressure instrument (0-300 psig, per LP85423, page 17), and is therefore not available for determining reactor pressure. B is incorrect – But is plausible to the Candidate who forgets that the other recorder is strictly a low-range pressure one. C and D are incorrect – But are plausible to the Candidate who, in addition to remembering the paperless recorders on P601, thinks there are also recorders on P870. This especially attractive to the ILT Candidate, where the Simulator uses a DCS Display screen on P870 to allow access to key plant Emergency Response parameters, including Reactor Pressure. This is for training, only; it is not the actual main control room configuration. Besides, this is a DCS Display computer point, and is not a recorder.

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

References provided to examinee:	None
References:	LP85423, Nuclear Boiler Instrumentation CPS 4200.01, Loss of AC Power

Date Written:	03/15/05	Author:	Ryder
Comments:			
There is no overlap between this question and any part of the Operating Exam. Although Candidates will have need to determine reactor pressure on a number of occasions in the Scenarios, especially, there is no particular time where they			

Question #	57

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295003 AA2.02	4.2	4.3	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of A.C. Power	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Reactor power, pressure, and level

will have to seek out a variety of other instruments, other than those usually-available (i.e., DCS displays), to do so.

Question #	58

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295021 AK3.04	3.3	3.4	Lower

System/Evolution Name:	Category:
Loss of Shutdown Cooling	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Maximizing reactor water cleanup flow

After a Loss of Shutdown Cooling, operators determine there is a NEED to MAXIMIZE the Bottom Head Drain Flow to RWCU, and to MAXIMIZE the cooling capability of RWCU.

Which ONE of the following describes the REASON for operating RWCU in this way?

A. To prevent excessive thermal stress of the Feedwater piping.

B. To minimize thermal stratification of the RPV.

C. To prevent erosion of the RPV bottom head drain line.

D. To minimize Feedwater line check valve flutter.

Answer: B

Explanation:

B is correct – Per CPS 3303.01, Sections 8.2.5 and 8.2.6. With the loss of shutdown cooling, the absence of forced flow through the vessel can result in thermal stratification. The Section 8.2.6 NOTE (center of page 58) advises operators to maximize RWCU cooling in order to promote sufficient natural circulation to minimize the concern for thermal stratification. The Section 8.2.5 NOTE (top of page 57) also advises operators to increase bottom head drain line flow to RWCU, to increase the forced (RWCU) circulation through the vessel and prevent bottom head region stratification.

A is incorrect – This choice is derived from another one of the several operational concerns related to RWCU. Specifically, CPS 3303.01, Section 6.4.2. Excessive feedwater line stress is prevented by limiting the delta-T between RWCU return temperature and Feedwater temperature when operating at low feedwater flow conditions. There is no cause-effect relationship between the way operators have decided to operate RWCU in the stem condition, and the limiting of this delta-T.

C is incorrect – This choice is derived from another RWCU operational concern. Specifically, CPS 3303.01, Section 6.10.1.1. When operators do maximize the bottom head drain flow (as suggested in the stem), they will necessarily have to comply with limiting that drain flow to 200 gpm, for the purpose of preventing drain line erosion. However, the REASON for operating RWCU in the way suggested is not directly related to this specific concern.

D is incorrect – Again, this choice is another operational concern. Specifically, CPS 3303.01, Section 6.11. Low flow conditions in the feedwater line can cause check valve flutter and excessive valve wear. Operators are advised to return all flows (RWCU and/or Shutdown Cooling) through a single line until there is sufficient flow (~ 300 gpm per line) to prevent the valve flutter. This, again, has no direct relationship to the REASON why the operators have decided to operate RWCU, as suggested in the stem.

Objective:	Question Source:	Level of Difficulty:
LP85204.1.2.6	New	2.9

References provided to examinee:	None
References:	CPS 3303.01, Reactor Water Cleanup System

Question #	58

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295021 AK3.04	3.3	3.4	Lower

System/Evolution Name:	Category:
Loss of Shutdown Cooling	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Maximizing reactor water cleanup flow

Date Written:	03/15/05	Author:	Ryder
Comments: None			

Question #	59

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295024 EK2.08	4.0	4.1	Lower

System/Evolution Name:	Category:
High Drywell Pressure	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: ADS

Assuming that all other logic permissives are satisfied, which ONE of the following identifies a COMBINATION of HIGH DRYWELL PRESSURE signals that will automatically initiate ADS?

Coincident signals from transmitters..

A. 'A' and 'C'

B. 'B' and 'D'

C. 'B' and 'F'

D. 'C' and 'E'

Answer: C

Explanation:
C is correct – Per LP85218, Figures 4 and 5. 'A' and 'E' will initiate ADS (Figure 4, Division 1 logic). 'B' and 'F' will initiate ADS (Figure 5, Division 2 logic). A, B, and D are incorrect – For the reasons described above.

Objective:	Question Source:	Level of Difficulty:
LP85218.1.4.1	New	2.8

References provided to examinee:	None
References:	LP85218, Automatic Depressurization System

Date Written:	03/16/05	Author:	Ryder
Comments: None			

Question #	60

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295030 EA1.06	3.4	3.4	Higher
System/Evolution Name:			**Category:**			
Low Suppression Pool Water Level			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (makeup to the suppression pool)						

Operators are implementing EOP-6, Primary Containment Control, with the following:

- Suppression Pool Water Level is rapidly lowering
- CRS directs the RO to makeup to the Suppression Pool via Cycled Condensate (CY), WITHOUT any assistance from operators in the field

Which ONE of the following:

(1) identifies how many valves the RO MUST OPEN to establish makeup flow,
and
(2) describes an additional action SUGGESTED by the procedure BEFORE opening the valve(s)?

A. (1) ONE valve
(2) START a second CY pump.

B. (1) TWO valves
(2) STOP the running CY pump.

C. (1) ONE valve
(2) STOP the running SF pump.

D. (1) TWO valves
(2) START a second SF pump.

Answer: A

Explanation:

A is correct – Refer to CPS 3220.01, Section 8.2. The RO needs only to open 1SM004 to establish makeup flow to the pool; one CY pump is normally running to provide the flow. The CAUTION advises that, "in an emergency, 1SM004 may be opened without throttling 1CY056". This valve (local) is normally wide open, and would first be fully closed (locally) before the RO opens 1SM004. The intent, here, is to provide CY pump run-out protection. For this question, (i.e., establishing makeup WITHOUT local operator assistance), however, the CAUTION suggests that first starting a second CY pump may help prevent CY pump run-out.

B is incorrect – For the reasons described above.

C and D are incorrect – Refer to *Section 8.3 of the procedure*. These choices are plausible to the Candidate who cannot recall the specific concern for establishing makeup flow in this 'emergency' mode. Section 8.3 utilizes Suppression Pool Cleanup and Transfer (SF) as an alterative (to using CY) pool makeup source. One SF pump (controlled at control room panel P870) is normally running; the Candidate may choose to stop it. A second SF pump is also available.

Question #	60

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	*1*	*1*	*295030 EA1.06*	3.4	3.4	Higher

System/Evolution Name:	**Category:**
Low Suppression Pool Water Level	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (makeup to the suppression pool)

Objective:	Question Source:	Level of Difficulty:
LP85408.1.2.2	New	2.1

References provided to examinee:	None
References:	CPS 3220.01, Suppression Pool Makeup (SM)

Date Written:	03/17/05	**Author:**	Ryder

Comments:

This question is categorized as Higher Cognitive (HCL) for the following reasons:

1. The pump runout concerns presented in this procedure section are not found in the Precuations/Limitations section of the procedure. If they were, then arguably this question would only demand the lowest level of mental processing (simple recall of information) and would be categorized as Lower Cognitive (LCL).

2. Candidates are not expected to have procedure sections, for evolutions such as this (i.e., important to protecting a lesser pump from a potential runout condition, but not a critical evolution vital to protecting the plant or to mitigating a significant transient), committed to memory.

3. As such, it is expected that most Candidates would have to use a higher level of mental processing to answer Part (2) of the question. This mental processing would be used to 'associate' concepts of potential impact on the system when opening the valves (i.e., recognition that this amounts to placing an flowpath burden on the running CY pump), with how this cause lead to pump runout, and then associating these pieces of information to derive a logical method for protecting the pump from runout.

4. Accordingly, without applying simple recall (and this being a closed-reference question), several 'Fundamentals' concepts have to be applied to answer Part (2).

Question #	61

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295037 EA2.06	4.0	4.1	Higher
System/Evolution Name:				**Category:**		
SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure						

Using the provided references, answer the following.

The plant is operating at rated power, when the main turbine trips, with the following:

- Operators enter EOP-1A for an ATWS and soon thereafter start BOTH SLC Pumps
- Operators are controlling reactor pressure between 800 and 1065 psig with the Turbine Bypass Valves and SRVs

Which ONE of the following describes a situation where it would be acceptable for operators to INTENTIONALLY LOWER the pressure control band?

A. The SLC Pumps have been running for 30 minutes.

B. Suppression Pool Temperature approaches 145°F.

C. Operators decide to use CD/CB to maintain the prescribed level band.

D. Operators decide to use low pressure ECCS to maintain the prescribed level band.

Answer: B

Explanation:

B is correct – Per EOP-6, Figure P (Heat Capacity Limit, HCL), as pool temperature nears 145°F (approximately), the HCL for a normal pool level (about 19 feet) is threatened. The HCL is considered one of the 'Critical Parameters' mentioned in CPS 4411.09, Section 4.3 (and 4.2). Even with an ATWS in progress, that procedure section allows operators to lower the pressure control band as necessary to stay within the HCL limit.

A is incorrect – CPS 4411.09, Section 4.2 prohibits intentionally lowering pressure (in an ATWS) until 'specific reactivity shutdown conditions have been established'. Refer to the PRESSURE leg of EOP-1A; specifically: the WAIT sign, and Table X. With both SLC pumps running for only '30 minutes' Cold Shutdown Boron has not yet been injected. Operators must remain at the WAIT sign for about 10 more minutes before intentionally lowering reactor pressure.

C and D are incorrect – These choices suggest that operators wish to lower pressure simply to get below the shutoff head of either Condensate/Condensate Booster Pumps (CD/CB) (~725 psig), or one or more of the low pressure ECCS pumps. CD/CB is a 'Preferred' ATWS injection system than might be used to maintain the prescribed level band in EOP-1A, LEVEL leg. Low pressure ECCS (LPCI/LPCS) is an 'Alternate' ATWS injection that might be used, only if level cannot be held above TAF (in this case, operators would be directed to blow down).

Question #	61

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295037 EA2.06	4.0	4.1	Higher

System/Evolution Name:	Category:
SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure

Objective:	Question Source:	Level of Difficulty:
LP87553.1.7.9	New	3.0

References provided to examinee:	EOP flowcharts
References:	CPS EOP-1A, ATWS RPV Control CPS EOP-6, Primary Containment Control CPS 4411.09, RPV Pressure Control Sources

Date Written:	03/17/05	Author:	Ryder
Comments: None			

Question #	62

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295038 EA2.01	3.3	4.3	Higher
System/Evolution Name:				**Category:**		
High Off-Site Release Rate				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site						

Per the Clinton Radiological Annex, which ONE of the following events WOULD BE considered an OFF-SITE radiological release?

A. A drum containing low-level radwaste falls off a truck and spills its contents on the ground JUST OUTSIDE the Protected Area fence; the spill is immediately contained, and does NOT get into the storm drainage system.

B. A temporary modification has MC cross-connected with CY when a rupture occurs on the MC Storage Tank; the spill from the tank reaches ONLY AS FAR AS the Nuclear Training Center, it does NOT get into the lake, and does NOT get into the storm drainage system.

C. The main stack effluent monitor alarms; field teams discover HIGHER than normal radiation levels ONLY AS FAR AS the railroad tracks that run along side of Highway 54, and ONLY in the direction of the 'AmerGen' sign.

D. Radiologically contaminated water is spilled into the lake on the intake structure side; grab samples show HIGHER than normal amounts of radionuclides, but due to dilution, the problem extends NO FARTHER THAN 500 YARDS from the shoreline, <u>and</u> there are NO radionuclides detected on the discharge side of the lake.

Answer: C

Explanation:
C is correct – Refer to EP-AA-1003, page CL-4, and to the CPS-ODCM, Figure 3.1-1. The perfect circle drawn around the 'Plant' is a 0.5 mile radius and is defined as the Site Boundary. Highway 54 is tangential to the edge of that Site Boundary. The site's 'Amergen' sign is at the end of the main access road, exactly at the point where the highway touches the Site Boundary edge. The railroad tracks (not shown on the Figure) run along side of the highway on the side <u>opposite</u> the plant. This choice describes a gaseous effluent release that has reached beyond the Site Boundary, and is an 'Off-Site' release. A is incorrect – This contained spill goes no further than perhaps 100 yards from the plant, and well within the Site Boundary. B is incorrect – The Nuclear Learning Center is on company property well within the Site Boundary, perhaps 400 yards from the MC Storage Tank. D is incorrect – The Site Boundary radius is 0.5 miles from the plant. 500 yards is about 0.28 miles. This liquid release is still 'on-site'.

Objective:	Question Source:	Level of Difficulty:
None	New	3.6

Question #	62

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295038 EA2.01	3.3	4.3	Higher

System/Evolution Name:	Category:
High Off-Site Release Rate	Emergency and Abnormal Plant Evolutions
KA Statement:	
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site	

References provided to examinee:	None
References:	CPS-Offsite Dose Calculation Manual (ODCM) EP-AA-1003, Clinton Radiological Annex

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	63

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	600000 AK1.02	2.9	3.1	Lower
System/Evolution Name:				**Category:**		
Plant Fire On-Site				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Knowledge of the operational implications of the following as they apply to PLANT FIRE ON-SITE: Fire fighting						

The plant is operating at rated power when the following occurs:

- Fire alarm (red flashing strobe light) occurs at MCR panel P661
- Operators CANNOT immediately CONFIRM whether or not an ACTUAL fire exists

Which ONE of the following describes the REQUIRED operator action?

A. Start a SECOND fire pump, so that TWO are running.

B. OPEN the FP CNMT and FP Drywell Isolation Valves.

C. Place ALL of the SRV control switches in OFF at P601.

D. MANUALLY initiate Halon in the main control room.

Answer: C

Explanation:
C is correct – Per CPS 1893.04, Section 8.1.2.1. P661 contains the Div 1 SRV logic. This action minimizes the risk that a fire in P661 could result in the auto-opening of the SRVs (plant depressurization) with 'A' solenoid (Div 1) control switches (OFF-AUTO-OPEN) still in AUTO. Refer to LP85239, pages 14-15. A is incorrect – Per CPS 1893.04, Section 8.1.7. Only one fire pump need be running. B is incorrect – Per CPS 1893.04, Section 8.1.8. Operators are directed to open these valves only when the fire is in CNMT or the drywell. D is incorrect – There is <u>no</u> requirement for initiating Halon unless an actual fire exists.

Objective:	Question Source:	Level of Difficulty:
None	New	2.8

References provided to examinee:	None
References:	LP85239, Main Steam System CPS 1893.04, Fire Fighting

Date Written:	03/19/05	**Author:**	Ryder
Comments: None			

Question #	64

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295032 EK3.03	3.8	3.9	Lower

System/Evolution Name:	Category:
High Secondary Containment Area Temperature	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: *Isolating affected systems*

Per EOP-8, Secondary Containment Control, which ONE of the following describes a situation where operators ARE PERMITTED to isolate the GIVEN plant system when its discharge IS THE CAUSE of a high area temperature?

A. RCIC, when it is needed for reactor pressure control in EOP-1.

B. RHR 'B', ANYTIME it is NOT needed for injection in EOP-1, EOP-1A, or EOP-2.

C. RWCU, when it is needed for reactor pressure control in EOP-1A.

D. MSL Drains, WHEN the RPV has been flooded to the Main Steam Lines in EOP-2.

Answer: D

Explanation:
D is correct – Per EOP-2, bottom-most WAIT steps. A and C are incorrect – Each of these describes a situation where a system is 'needed' for EOP actions. Isolating such a system is <u>not</u> allowed by EOP-8 (step 31). B is incorrect – This choice is incorrect because the word 'anytime' suggests that since the system is not needed for injection (adequate core cooling), operators <u>could</u> isolate it if it were needed for Containment Spray in EOP-6. This is <u>not</u> true.

Objective:	Question Source:	Level of Difficulty:
LP87559.1.2.2	New	3.7

References provided to examinee:	None; access to EOP flowcharts is OK
References:	CPS EOP-1, RPV Control CPS EOP-1A, ATWS RPV Control CPS EOP-2, RPV Flooding CPS EOP-6, Primary Containment Control CPS EOP-8, Secondary Containment Control

Date Written:	05/16/05	Author:	Ryder

Comments:

Given this KA, we have taken an 'applications' approach as a means of creating a question that provides sufficient discrimination. To correctly answer this question, the Candidate must correctly apply the requirements of EOP-8, step 31. The fact that the WAIT steps at the bottom of EOP-2 <u>require</u> operators to isolate the MSL Drains does <u>not</u> detract from the fact that doing so is also consistent with the EOP-8, step 31, allowance.

Question #	65

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	205000 A1.05	3.4	3.4	Lower

System/Evolution Name:	Category:
Shutdown Cooling System (RHR Shutdown Cooling mode)	Plant Systems

KA Statement:
Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM/MODE controls including: Reactor water level

With REACTOR PRESSURE LESS THAN 30 PSIG, operators are warming RHR loop 'B' in preparation for placing it in Shutdown Cooling.

Per CPS 3312.03, Shutdown Cooling, which ONE of the following describes a POTENTIAL plant/system response when operators INITIATE WARMUP FLOW.

A. Reactor water level SUDDENLY UNCONTROLLABLY LOWERS.

B. Reactor pressure SUDDENLY UNCONTROLLABLY RISES.

C. RHR Pump RAPIDLY approaches RUNOUT.

D. RHR Pump RAPIDLY OVERHEATS.

Answer: A

Explanation:
A is correct – Per CPS 3312.03, Section 5.4. At such a low pressure (<30 psig), there is a chance that warmup flow (i.e., RHR Pump is not yet running) may not be sufficient to open the RHR Pump Discharge Check Valve, F031. This could cause the downstream piping to drain to radwaste when operators open the radwaste discharge valves. If the pipe volume empties, and then the F031 opens (likely, due to the downstream pressure now being relieved), a sudden, uncontrollable drop in reactor water level (level transient) will occur as that piping refills. See LP85205, Figure 5. B is incorrect – This choice has face validity, but is not a concern during this evolution. C and D are incorrect – Candidate is expected to know that the RHR Pump is not yet running when 'warmup flow' is initiated.

Objective:	Question Source:	Level of Difficulty:
LP85205.1.14	New	3.0

References provided to examinee:	None
References:	CPS 3312.03, RHR Shutdown Cooling LP85205, RHR

Date Written:	03/19/05	Author:	Ryder
Comments: None			

Question #	66

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	215003 K1.06	3.9	4.0	Higher

System/Evolution Name:	Category:
Intermediate Range Monitor (IRM) System	Plant Systems

KA Statement:
Knowledge of the physical connections and/or cause-effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: APRM SCRAM signals

The plant is in MODE 2, with the following:

- Plant heatup and pressurization is in progress
- RO is ranging up the IRMs, AS REQUIRED BY PROCEDURE

Which ONE of the following identifies the LOWEST IRM READING (DCS display) where the APRMs could POSSIBLY provide the PRIMARY trip signal for a core-wide rise in reactor power?

An IRM reading of...

A. 70 on Range 9

B. 90 on Range 9

C. 20 on Range 10

D. 40 on Range 10

Answer: D

Explanation:
D is correct – Per LP85411, pages 24, 25, and Figure 14, and per CPS 3306.01, Section 6.4. In MODE 2, the Mode Switch is not in RUN. The APRM Upscale Neutron Trip setpoint is 15%. An IRM Range 10 reading of 40 is within instrument loop tolerance for this trip signal, at about 13% APRM power. A and C are incorrect - These equate to Range 10 readings of about 23 and 30, respectively, and are about 7% and 10% APRM power, respectively. B is incorrect – Of the distracters, this equates to the 2nd highest APRM power level (about 10%). Even if it were shown that the 15% APRM upscale trip instrument loop could reach down as far as 10%, this choice is incorrect because, per CPS 3306.01, Section 6.4, the RO should have ranged up <u>before</u> reaching this '90' value on Range 9. Stem conditions are explicit about the fact that operators ARE ranging up 'as required by procedures'.

Objective:	Question Source:	Level of Difficulty:
LP84411.1.11.1	New	3.0

References provided to examinee:	None
References:	LP85411, APRM/LPRM System CPS 3306.01, Source/Intermediate Range Monitors

Date Written:	03/21/05	Author:	Ryder
Comments: None			

Question #	67

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	2	295017 AA2.03	3.1	3.9	Lower

System/Evolution Name:	Category:
High Off-Site Release Rate	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels

The plant is operating at power when a VALID, ALERT level alarm occurs on the in-service HVAC Stack Effluent Monitor.

Which ONE of the following describes the operational and/or radiological significance of this alarm?

A. A Technical Specification entry is required.

B. An EOP-9, Radioactivity Release Control, entry is required.

C. NEITHER the ON-SITE, NOR the OFF-SITE exposure limits are being exceeded.

D. The ON-SITE exposure limits MIGHT have been exceeded; the OFF-SITE exposure limits are NOT being exceeded.

Answer: C

Explanation:
C is correct – Per CPS 5140.41, and CPS 4979.01, Section 6.4. NO exposure limits are exceeded when the PRM alarm level is only at the ALERT point. There are 'several decades of buffer' between this alarm point and the on-site and off-site exposure limits. A is incorrect – No Tech Spec entry is required for this 'valid' alarm. Choice offers face validity and plausibility based on Candidate attraction to answer choices that suggest a recall of 'less than one hour' Tech Spec actions. B is incorrect – This choice is attracting to the Candidate who confuses the ALERT alarm point of a process radiation monitor channel with the 'ALERT' Emergency Action Level (EAL) associated with an EOP-9 entry condition. D is incorrect – For the reason associated with the correct answer, 'C'.

Objective:	Question Source:	Level of Difficulty:
None	New	3.1

References provided to examinee:	None
References:	CPS EOP-9, Radioactivity Release Control CPS 5140.41, alarm response for 0RIX-PR001 CPS 4979.01, Abnormal Release of Airborne Radioactivity

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	68

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295018 AA1.03	3.3	3.4	Higher
System/Evolution Name:				**Category:**		
Partial or Complete Loss of Component Cooling Water				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Affected systems so as to isolate damaged portions						

The plant is operating at power when the following occurs:

- CCW Effluent Monitor, 1RIX-PR037, alarms (a valid alarm)
- 1RIX-PR037 is nearing its HIGH alarm setpoint
- Source of the alarm is a tube leak on the ONLY available NRHX
- Leak rate is about 29 gpm

Which ONE of the following describes the required operator action?

A. Enter CPS 4001.02, Automatic Isolation.

B. Isolate CCW from the NRHX and open the RWCU Heat Exchanger Bypass, 1G33-F104.

C. Stop the RWCU Pumps and isolate the RWCU system.

D. Commence a normal plant shutdown within 1 hour, and be in MODE 3 within 12 hours.

Answer: C

Explanation:

C is correct – Per CPS 5140.49, Operator Action #1, and per CPS 3303.01, Section 8.3.3. Operation of RWCU with the plant above 120°F and CCW isolated from the NRHX cannot continue. Operators are directed to remove RWCU from service.

A is incorrect – Per CPS 4001.02C001, page 3. RWCU delta-flow isolation setpoint is 59 gpm after 45 seconds. Per Computer Point E31DA001, the normal, at power, sensed RWCU differential flow (due to instrument calibration impact by system flow dynamics) is about 25 gpm. The system should still be un-isolated with the tube leaking at only 29 gpm; total system delta-flow is well below the 59 gpm isolation setpoint (i.e., about 54 gpm). There is no reason for operators to enter the Automatic Isolation procedure.

B is incorrect – For the reason associated with the correct answer, 'C'.

D is incorrect – This choice suggests that Tech Spec 3.0.3 applies. It does not. Its plausibility is rooted in its attraction to the Candidate who wants to consider the RCS Leakage aspect of this event. This NRHX tube leak does <u>not</u> constitute a RCS Pressure Boundary leak (defined in Tech Spec 1.1); therefore, Tech Spec 3.4.5.C does not apply here. Although this Tech Spec is not a 3.0.3 entry, this choice is worded in a way that suggests the need for a Candidate to recall short-term (< 1-hour) Tech Specs from memory.

NOTE: Each of these choices suggests that operators have determined the need to isolate the source of the release. The stem avoids explicitly suggesting this in order to make choices 'A' and 'D' plausible. Tech Spec 3.4.5.A, alone, is sufficient to argue that isolation of the leak is in fact required.

Question #	68

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	1	1	295018 AA1.03	3.3	3.4	Higher

System/Evolution Name:	Category:
Partial or Complete Loss of Component Cooling Water	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Affected systems so as to isolate damaged portions

Objective:	Question Source:	Level of Difficulty:
LP85204.1.12	New	2.6

References provided to examinee:	None
References:	CPS 3303.01, Reactor Water Cleanup System CPS 4001.01, Automatic Isolation CPS 4001.02C001, Automatic Isolation Checklist CPS Tech Spec 3.4.5, RCS Operational Leakage CPS Tech Spec 1.1, Definitions CPS 5140.49, 1RIX-PR037, alarm response procedure

Date Written:	03/21/05	Author:	Ryder
Comments: None			

Question #	71

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	223002 2.4.31	3.3	3.4	Lower

System/Evolution Name:	Category:
PCIS/Nuclear Steam Supply Shutoff System	Plant Systems

KA Statement:
Knowledge of annunciator alarms and indications, and use of the response procedures

Which ONE of the following identifies an annunciator for which the alarm response procedure directs operators to place the Div 1 Sensor Bypass Switch in BYPASS if the alarming (tripped) condition CANNOT otherwise be removed?

A. 5067-6D, DIV 1 TRIP UNIT OUT OF FILE

B. 5067-1H, INBOARD LOSS OF ISOLATOR POWER

C. 5067-7B, LDS P632 ISOLATOR CARD POWER LOSS

D. 5063-8A, DIV 1 SAFETY ASSOCIATED ATM TROUBLE

Answer: A

Explanation:

A is correct – Per CPS 5067-6D, Operator Action 2.a. Although the Candidate is not expected to know this action from memory, he/she is expected to recognize that this annunciator alarms whenever any one of 9 separate analog trip modules (ATMs) becomes unseated (dislodged) in its rack, and that this produces a Div 1 trip for the associated parameter (e.g., a Group 1 isolation half-trip is generated if 1B21-N681A (RPV Level 1) is the offending ATM).

B is incorrect – Per CPS 5067-1H. This annunciator is associated solely with the MSIV Leakage Control System, and is not affiliated with the Div1 NSPS logic cabinet (P661) or its components, including the Sensor Bypass Switch.

C is incorrect – Per CPS 5067-7B. This belongs to the Div 1 portion of Leak Detection (LDS) and is not affiliated with the Div1 NSPS logic cabinet (P661) or its components, including the Sensor Bypass Switch.

D is incorrect – Per CPS 5063-8A. This belongs to ATMs associated solely with RCIC is not affiliated with the Div1 NSPS logic cabinet (P661) or its components, including the Sensor Bypass Switch.

Objective:	Question Source:	Level of Difficulty:
LP85434.1.4.10	New	4.0

References provided to examinee:	None
References:	CPS 5067-6D, 1H, and 7B, and CPS 5063-8A, alarm response procedures

Date Written:	05/03/05	Author:	Ryder
Comments: None			

Question #	72

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.1.30	3.9	3.4	Higher
System/Evolution Name:			**Category:**			
			Conduct of Operations			
KA Statement:						
Ability to locate and operate components, including local controls						

The plant is in MODE 3, cooling down, when the following occurs:

- Low feedwater flow conditions cause annunciator 5000-2F, RWCU HI DIFF FLOW TIMER INITIATED, to alarm
- CRS decides to prevent an unnecessary RWCU isolation

Which ONE of the following describes:

(1) how procedures direct operators to PREVENT this RWCU isolation,
<u>and</u>
(2) the automatic response of RWCU, if operators are able to DEFEAT ONLY ONE division before the Differential Flow Timers time out?

A. (1) At P632 <u>and</u> P642, manually dial the associated Differential Flow Timer fully COUNTER-CLOCKWISE.
(2) RWCU isolates.

B. (1) At P855, manually dial BOTH Differential Flow Timers fully COUNTER-CLOCKWISE.
(2) RWCU does NOT isolate.

C. (1) At P632 <u>and</u> P642, place the associated RWCU Isolation Bypass switch in BYPASS.
(2) RWCU isolates.

D. (1) At P855, place BOTH RWCU Isolation Bypass switches in BYPASS.
(2) RWCU does NOT isolate.

Answer: C

Explanation:

C is correct – Refer to CPS 5000-2F, LP85404, page 25, and LP85204, page 16. Part (1) – The Div 1 RWCU Isolation Bypass switch is located on P632 (Div 1 Leak Detection System panel); similarly, the Div 2 switch is on P642 (Div 2). At each of these panels, P632 and P642, is also that division's Diff-Flow Timer (referred to in CPS 5000-2F as 1E31-R615A and B). Although it is physically possible to manually dial back a running timer, it is not the method prescribed by procedures, including CPS 5000-2F. Part (2) – A given Isolation Bypass switch defeats the isolation only for the isolation valves controlled by that division. The Outboard isolations are Div 1, the Inboards are Div 2. Even if operators only manage to place one of the switches in BYPASS before 45 seconds has expired (the Timer setpoint), the valves controlled by the opposite Division will close. Thus, RWCU will still isolate.

A, B, and D are incorrect – For the reasons described above.

Question #	72

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.1.30	3.9	3.4	Higher

System/Evolution Name:	Category:
	Conduct of Operations

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

Objective:	Question Source:	Level of Difficulty:
LP85404.1.11.26	New	2.2

References provided to examinee:	None
References:	LP85204, Reactor Water Cleanup System LP85404, Leak Detection System CPS 5000-2F, RWCU Hi Diff Flow Timer Initiated, alarm response

Date Written:	03/22/05	Author:	Ryder

Comments:

These panels, P632 and P642, are within the control room complex, but <u>not</u> within the 'control board' area of where the operators are normally stationed. That is, they are 'back panels'. These 'back panels' satisfy the intent of this Generic KA. Arguably, this KA does not <u>necessarily</u> demand that the question specifically address 'local controls', but rather that it <u>may</u> address local controls <u>or</u> MCR controls.

Question #	73

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	3	Generics	2.4.43	2.8	3.5	Lower
System/Evolution Name:			**Category:**			
			Emergency Procedures and Plan			
KA Statement:						
Knowledge of emergency communications systems and techniques						

Which ONE of the following identifies operational features available at the Remote Shutdown Panel area?

A. Can initiate the ALL PAGE mode for the Gaitronics system, and sound the plant GENERAL PURPOSE alarm.

B. Can sound the CONTAINMENT EVACUATION alarm, and sound the FUEL BUILDING EVACUATION alarm.

C. Can MANUALLY operate TWO of the ADS-SRVs.

D. Can inject to the RPV with LPCI Loop 'C'.

Answer: A

Explanation:
A is correct – Per CPS 1021.01, Section 8.1.4. ALL CALL feature is available, and so is any plant alarm than can be manually initiated. Per CPS 3842.01, Section 8.1.2, this feature known by how the associated pushbutton is actually labeled: 'All Page'. B is incorrect – Per CPS 1021.01, Section 8.1.4. The Fuel Building Evacuation alarm can only be automatically initiated (by radiation monitoring); it has no manual feature, either at the RSP or in the MCR. C is incorrect – Per LP85433, page 16. Only one ADS-SRV (F051G) can be manually operated at the RSP. D is incorrect – Per LP85433, page 6. Although LPCI Loop 'C' <u>is</u> a Div 2 subsystem, it cannot be operated to inject from the Remote Shutdown Panel area. The other Div 2 LPCI Loop, 'B', <u>can</u> be used.

Objective:	Question Source:	Level of Difficulty:
None	New	2.2

References provided to examinee:	None
References:	*CPS 1021.01, Site Communications* CPS 3842.01, Plant Communications Alarm Test LP85433, Remote Shutdown

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	74

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	300000 A4.01	2.6	2.7	Higher

System/Evolution Name:	Category:
Instrument Air System (IAS)	Plant Systems

KA Statement:
Ability to manually operate and/or monitor in the control room: Pressure gauges

The SERVICE AIR HEADER PRESSURE gauge on P800 has three colored bands: Green, Yellow, and Red.

WITHOUT operator action, which ONE of the following describes the expected status of Plant Air components ONE MINUTE after an operator sees the reading on the gauge drop into the RED band because of a significant air leak?

A. TWO service air compressors are running; one or more automatic ring header isolations are CLOSED.

B. ONLY ONE service air compressor is running; all automatic ring header isolations are OPEN.

C. TWO service air compressors are running; all automatic ring header isolations are OPEN.

D. ALL THREE service air compressors are running; one or more automatic ring header isolations are CLOSED.

Answer: A

Explanation:
A is correct – The RED band covers pressures at or below 70 psig (observed in both the MCR and Simulator configurations). Per CPS 5041-6B, the standby air compressor should have auto-started at 80 psig (lowering); therefore, two are now running. Per CPS 5041-5C, the ring header isolations should close at 70 psig (lowering). The stem uses 'one minute' as a way to ensure that all portions of the Plant Air system have had sufficient time to sense the lowered pressure, and to ensure that all affected ring header auto isolations have closed as a result. B and C are incorrect – For the reasons described above. D is incorrect – The 3rd air compressor is normally in Pull-To-Lock (PTL) and will <u>not</u>, therefore, auto-start.

Objective:	Question Source:	Level of Difficulty:
LP85301.1.7	New	3.0

References provided to examinee:	None
References:	CPS 5041-5C, and –6B alarm response procedures

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	75

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	300000 K5.01	2.5	2.5	Higher

System/Evolution Name:	Category:
Instrument Air System (IAS)	Plant Systems

KA Statement:
Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors

A STATION BLACKOUT is in progress.

Per CPS 4200.01, Loss of AC, which ONE of the following describes what operators are directed to do with the PLANT AIR system, and why?

A. In the main control room, place all 3 Service Air Compressor control switches in Pull-To-Lock, to prevent their auto-restart (when power is restored) until their support systems are also made available.

B. In the field, place all 3 Service Air Compressor Mode Selector switches in the UNLOAD position, to prevent auto-restart of the compressors (when power is restored) until their support systems are also made available.

C. In the main control room, place the Containment SERVICE Air Header Isolation Valves control switches in the CLOSE position, to preserve available air for vital plant equipment systems, when plant air pressure is restored.

D. In the field, gag SHUT the SERVICE Air Ring Header Isolation Valves for the Radwaste, Turbine, Control, and Auxiliary Buildings, to preserve available air for vital plant equipment, when plant air pressure is restored.

Answer: A

Explanation:
A is correct – Per CPS 3214.01, Section 2.1.a, and CPS 4200.01, Appendix A. No further explanation required. B is incorrect – Per CPS 3214.01, Section 6.5. The only way to protect the compressors on power restoration is by placing them in PTL. Even if the in-field switches are placed in UNLOAD, the compressors can still auto-start (unless in PTL) and run unloaded. C is incorrect – Refer to LP85301, page 24. These are 3-position (CLOSE-AUTO-OPEN) control switches, normally in AUTO. They will in fact auto-reopen when air pressure is restored, but that's OK. There are no requirements to prevent these valves from reopening in an effort to 'preserve' air (especially Instrument Air) for more important plant equipment. D is incorrect – For the same reason associated with choice 'C'. Also, these valves CANNOT auto-reopen. They trip shut on low air header pressure (70 psig) and must be manually re-latched and re-opened in the field. There is no need to 'gag' them shut. Refer to LP85301, pages 22-23.

Objective:	Question Source:	Level of Difficulty:
PB420001.1.3.3	New	3.0

Question #	75

RO/SRO:	Tier:	Group:	KA:	RO IR:	SRO IR:	Cog Level
Both	2	1	300000 K5.01	2.5	2.5	Higher

System/Evolution Name:	**Category:**
Instrument Air System (IAS)	Plant Systems
KA Statement:	
Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors	

References provided to examinee:	None
References:	LP85301, Service Air and Instrument Air CPS 3214.01, Plant Air CPS 4200.01, Loss of AC

Date Written:	04/29/05	**Author:**	Ryder
Comments: None			

Question #	76

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295004 2.2.25	55.43(b)(2)	3.7	Higher
System/Evolution Name:			Category:			
Partial or Complete Loss of DC Power			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of bases in technical specifications for limiting conditions for operations and safety limits						

Using the provided references, answer the following.

The plant is operating at rated power when the following occurs:

- At Time = 0000 hours, a fault in the RAT supply breaker causes 4160V Bus 1C1 to transfer to the ERAT
- At Time = 0300 hours, a fault in the RAT supply breaker causes 4160V Bus 1B1 to transfer to the ERAT
- At Time = 1500 hours, the supply breaker to Div 1 125 VDC Distribution Panel, 1A, trips open and will NOT re-close

If NONE of these failures can be corrected, which ONE of the following identifies the LATEST time by when the plant MUST be in MODE 3?

The plant must be in MODE 3 no later than...

A. 13 hours after the DC supply breaker fails.

B. 14 hours after the DC supply breaker fails.

C. 36 hours after the RAT breaker for Bus 1B1 fails.

D. 84 hours after the RAT breaker for Bus 1C1 fails.

Answer: B

Explanation:
B is correct – Per TS 3.8.9, Conditions C and D. For Condition C, only the '2 hour' completion time applies. Per Condition D, the plant must be in Mode 3 within 14 hours (2-hr allowed outage time + 12 hours completion time = 14 hours) after the DC supply breaker failure. To avoid applying the '16 hours from discovery…' completion time of Condition C, the SRO Candidate must recognize/recall the following portions of TS 3.8.9 Bases: 1) on page B 3.8-78, the RAT breaker failures for the AC buses do not constitute 'Distribution System' inoperabilities (because the ERAT source is still available to the buses); 2) once this is recognized, then the Candidate will avoid looking at these failures (both AC buses, followed by the DC panel) as a string of 'contiguous' failures which would otherwise require the application of the '16 hours' constraint. A is incorrect – See the explanation above. The Candidate will choose this as the answer if he/she inappropriately applies the '16 hours…' completion time of TS 3.8.9, Condition C. C is incorrect – This is not correct because the DC breaker problem is more limiting (as described above). It is plausible to the Candidate who inappropriately applies TS 3.8.1, Condition C; i.e., thinks the loss of the RAT supply (<u>one</u> offsite circuit) to <u>two</u> divisional buses (1B1 and 1C1) is synonymous with the loss of 'two offsite circuits'.

Question #	76

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295004 2.2.25	55.43(b)(2)	3.7	Higher
System/Evolution Name:				**Category:**		
Partial or Complete Loss of DC Power				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Knowledge of bases in technical specifications for limiting conditions for operations and safety limits						

D is incorrect – Per TS 3.8.1, Condition A, then Condition F. Were it not for the more limiting DC distribution problem, this would be the correct answer.

Objective:	Question Source:	Level of Difficulty:
LP85263.1.16	New	2.3

References provided to examinee:	CPS Tech Spec Section 3.8 (without Bases)
References:	CPS TS 3.8.1, AC Sources – Operating CPS TS 3.8.9, Distribution Systems – Operating (and Bases)

Date Written:	04/29/05	**Author:**	Ryder
Comments: None			

Question #	77

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295016 2.4.6	55.43(b)(5)	4.0	Higher
System/Evolution Name:			**Category:**			
Control Room Abandonment			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Knowledge of symptom based EOP mitigation strategies						

Using the provided references, answer the following.

The plant is operating at rated power when the following occurs:

- Shift Manager determines that a control room evacuation is required
- Before leaving, operators place the Mode Switch in SHUTDOWN
- An ATWS results
- Operators arm and depress the Manual Scram pushbuttons and manually initiate ARI
- The SLC Pumps will NOT start
- Personnel abandon the control room with the following:
  - Main turbine is on line
  - Reactor power is 35%
  - Scram air header is DE-PRESSURIZED
  - Main control room is UNINHABITABLE and INACCESSIBLE

Which ONE of the following describes the NEXT appropriate operator action?

A. Locally open the SLC Storage Tank Outlets and start the SLC Pumps.

B. Scram all control rods using the HCU Scram Test Switches.

C. Terminate and prevent Feedwater, HPCS, and RCIC.

D. Defeat the MSL/OG and IA Interlocks.

Answer: C

Explanation:

C is correct – Per CPS EOP-1A, Level leg, and CPS 4003.01, Sections 4.3 and 4.4. Operators <u>do</u> have the facilities to prevent HPCS injection (at the Div 3 switchgear, per Section 4.4.4), Feedwater injection (by closing the Feedwater Shutoffs per Section 4.4.3), and RCIC (controllable at the Remote Shutdown Panel). The objective here would be to lower RPV water level to –60 inches and establish Level Band 'B'.

A is incorrect – Besides there being no procedure guidance for this, the SLC squib valves will <u>not</u> fire when the pumps are started locally (see LP85211, page 21).

B is incorrect – This is allowed by CPS EOP-1A, Power leg, and CPS 4411.08, Section 2.6. However, this is time intensive and would be a vain attempt to solve the ATWS problem, given that the stem conditions indicate a 'hydraulic' type of ATWS exists (i.e., scram air header has already de-pressurized). This is <u>not</u> the NEXT appropriate action.

D is incorrect – Per CPS EOP-1A, Level leg, and CPS 4410.00C004, this requires accessibility to several main control room panels. Stem conditions indicate the control room is <u>not</u> accessible.

Question #	77

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295016 2.4.6	55.43(b)(5)	4.0	Higher

System/Evolution Name:	Category:
Control Room Abandonment	Emergency and Abnormal Plant Evolutions

KA Statement:
Knowledge of symptom based EOP mitigation strategies

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	EOP flowcharts
References:	CPS 4003.01, Remote Shutdown CPS EOP-1A, ATWS RPV Control CPS 4411.08, Alternate Control Rod Insertion CPS 4410.00C004, Defeating MSL/OG Interlocks LP85211, Standby Liquid Control

Date Written:	03/29/05	Author:	Ryder
Comments: None			

Question #	78

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295025 EA2.01	55.43(b)(2)	4.3	Higher
System/Evolution Name:				**Category:**		
High Reactor Pressure				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure						

Multiple system failures have resulted in Reactor Pressure rising to a PEAK of 1340 psig, as indicated on control room recorders.

Which ONE of the following identifies:

(1) the Reactor Coolant System (RCS) portion MOST impacted by this overpressure transient,
<u>and</u>
(2) whether or not that RCS portion's MAXIMUM ALLOWED TRANSIENT PRESSURE value has been EXCEEDED?

A. (1) Recirc pump DISCHARGE piping
(2) Has NOT been exceeded.

B. (1) Recirc pump SUCTION piping
(2) HAS been exceeded.

C. (1) RPV BOTTOM Head
(2) Has NOT been exceeded.

D. (1) RPV TOP Head
(2) HAS been exceeded.

Answer: B

Explanation:

B is correct – Refer to Tech Spec SL 2.1.2 Basis for all of the answer choices. The RCS suction piping is at the lowest elevation of any RCS portion. The 1325 psig (steam dome) SL value equates to 1375 psig at the lowest elevation portion of the RCS (i.e., Recirc suction piping); i.e. a +50 psig difference. Therefore, an overpressure transient peak pressure of 1340 psig (on the control room recorders, which look at steam dome pressure) equates to 1390 psig in the Recirc suction piping. The 'maximum allowed transient pressure value' for any portion of the RCS is: 110% of the <u>Design</u> pressure value for that portion. The Design pressure value for the Recirc suction piping is 1250 psig. Therefore, the 'max allowed transient pressure value' is 1375 psig (1.1 x 1250 = 1375). Therefore, the overpressure of 1390 psig in the Recirc suction piping portion of RCS <u>does</u> exceed this 'max allowed…' value, and this RCS portion is clearly the 'most' impacted, relative to the other given RCS choices.

A is incorrect – The Design pressure for the Recirc discharge piping is 1550 psig or 1650 psig, depending on the location relative to the discharge valve. As such, these portions are <u>not</u> the 'most' impacted. Additionally, the 110% values are 1705 psig and 1815 psig, respectively. Since these portions are at elevations higher than the Recirc piping, the delta-pressure between these portions and the RPV steam dome is something less than +50 psig. Therefore, the 1340 psig steam dome overpressure transient equates Recirc discharge piping pressures that far below the respective 'max allowed…' values.

C and D are incorrect – A 1340 psig steam dome value equates to that same value throughout the RPV. In fact, the 'steam dome' is essentially synonymous with 'top head'. The RPV Design pressure is 1250 psig. Its 'max allowed'

Question #	78

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295025 EA2.01	55.43(b)(2)	4.3	Higher

System/Evolution Name:	Category:
High Reactor Pressure	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure

(110%) value is 1375 psig. The 1340 psig overpressure transient has not exceeded that value.

Objective:	Question Source:	Level of Difficulty:
LP87621.1.6	New	2.3

References provided to examinee:	None
References:	CPS Tech Spec SL 2.1.2, Reactor Coolant System Pressure SL (and Basis)

Date Written:	03/30/05	Author:	Ryder
Comments: None			

Question #	79

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295021 AA2.01	55.43(b)(6)	3.6	Higher
System/Evolution Name:				**Category:**		
Loss of Shutdown Cooling				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water heatup/cooldown rate						

Using the provided references, answer the following.

The plant is in MODE 4, making preparations for refueling, with the following:

- The reactor was shut down 6 days ago
- Reactor water temperature is 122°F

- THEN, a complete Loss of Shutdown Cooling occurs
- NO Reactor Recirc Pump is available
- There is NO readily available means of restoring shutdown cooling

If operators were to MAXIMIZE the reactor water level, as allowed by procedures, how long will it take before a Mode change is required?

A. 67 minutes

B. 90 minutes

C. 102 minutes

D. 120 minutes

Answer: C

Explanation:
C is correct – Per CPS 4006.01, Section 4.6.6, operators are allowed to raise RPV water level as high as the main steam lines to gain an initial cooling effect and delay a Mode 3 entry. Section 4.6.4 directs operators to refer to the Heatup Rate and Boil-off Time Curves to assess the heatup rate that can be expected. The SRO Candidate should review the curve labeled 'Reactor Vessel Heatup Rate – Before Refueling' and plot a "°F/Hr" point where '6 days after shutdown' intersects the 'Main Steam Lines' water level curve. The result of this plot yields an approximate 46°F/hr heatup rate. With current reactor water temperature at 122°F, a Mode 3 entry (200°F) is 78°F away. From this point a simple calculation shows that it will take about 1.7 hours, or 102 minutes, to reach 200°F. Calculation: 78 ÷ 46 = 1.7 = 1 hour, 42 minutes = 102 minutes. A is incorrect – This choice is plausible to the Candidate who carelessly translates 1.7 hours (as defined above) to 1 hour, 7 minutes (67 minutes). Its plausibility is based on a demonstrated propensity for people to make exactly this kind of careless mistake. B is incorrect – Refer to the same explanation as for the correct answer, 'C'. This is the calculated time if the Candidate were to believe that the maximum allowed RPV water level is +44 inches Shutdown Range. The idea that the Candidate might believe this is soundly based on the fact that Candidates readily associate +44 inches with the minimum level necessary to promote adequate natural circulation in the absence of forced cooling flow. SRO

Question #	79

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295021 AA2.01	55.43(b)(6)	3.6	Higher

System/Evolution Name:	Category:
Loss of Shutdown Cooling	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to *LOSS OF SHUTDOWN COOLING*: Reactor water heatup/cooldown rate

Candidate is expected to recall that CPS 4006.01, Section 4.6.6 allows operators to raise level as high as the main steam lines. Calculation: 78 + 52 = 1.5 hours = 1 hour, 30 minutes = 90 minutes.

D is incorrect – Similarly, this is the calculated result if the Candidate inappropriately applies the 'Vessel Flange' water level curve. This level is higher than allowed by CPS 4006.01, Section 4.6.6. Calculation: 78 ÷ 39 = 2 hours = 120 minutes.

Objective:	*Question Source:*	Level of Difficulty:
None	New	3.0

References provided to examinee:	The 'Reactor Vessel Heatup Rate – Before Refueling' curve, discussed above
References:	CPS 4006.01, Loss of Shutdown Cooling 'Reactor Vessel Heatup Rate – Before Refueling' curve, discussed above

Date Written:	03/31/05	Author:	Ryder
Comments: None			

Question #	80

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	600000 AA2.09	55.43(b)(5)	2.8	Higher
System/Evolution Name:				**Category:**		
Plant Fire On Site				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: That a failed fire alarm detector exists						

Using the provided references, answer the following.

The plant is operating at rated power when TWO of the smoke detectors in Fire Zone F-1b are determined to be INOPERABLE.

Which ONE of the following describes the required action?

A. Restore BOTH of these detectors to an OPERABLE status within 14 days; otherwise, establish a fire watch to inspect the zone hourly.

B. Restore AT LEAST ONE of these detectors to an OPERABLE status within 14 days; otherwise, have a fire watch inspect the zone hourly, thereafter.

C. AFTER declaring these detectors inoperable, within 1 hour establish a fire watch to inspect the zone hourly.

D. AFTER declaring these detectors inoperable, within 1 hour inspect the zone, and inspect it hourly, thereafter.

Answer: C

Explanation:
C is correct – Per CPS 1893.01, Appendix D, page 37, TWO inoperable detectors in this zone constitutes having 'more than half' of the 3 total detectors, that are in this zone, inoperable. This being a HPCS equipment zone, these detectors are required to be OPERABLE, because HPCS is required to OPERABLE at rated power (Mode 1, per Tech Spec 3.5.1). Per CPS 1893.01, Appendix A, page 21, fire protection impairment Compensatory Measure 9.b applies for this case. Within 1 hour of declaring the detectors inoperable, a fire watch must be established, then hourly inspections of zone F-1b must commence. A is incorrect – This would be the required action if there were at least 4 total detectors in fire zone F-1b. This is the action directed by Compensatory Measure 9.a, on page 20. B is incorrect – This choice has face validity and is plausible based on two premises: 1) The very difficult-to-read Compensatory Measure 9.a making the Candidate vulnerable to misreading the requirements, and 2) an operability restoration technique very often employed within Tech Specs; i.e., the idea that as soon as at least one of the 2 detectors can be restored to operability, the Compensatory Measure can be exited, and no further action would be required. This is <u>not</u> the case. D is incorrect – This choice is essentially how a Candidate could easily <u>misread</u> the very difficult-to-read Compensatory Measure 9.b.

Question #	80

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	600000 AA2.09	55.43(b)(5)	2.8	Higher

System/Evolution Name:	*Category:*
Plant Fire On Site	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: That a failed fire alarm detector exists

Objective:	Question Source:	Level of Difficulty:
None	New	2.2

References provided to examinee:	CPS 1893.01, in its entirety
References:	CPS 1893.01, Fire Protection Impairment Reporting

Date Written:	03/31/05	Author:	Ryder
Comments: None			

Question #	81

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	2	295010 AA2.02	55.43(b)(2)	3.9	Higher
System/Evolution Name:			**Category:**			
High Drywell Pressure			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell pressure						

The plant is operating at rated power, when the following occurs:

- A PARTIAL loss of Drywell Cooling (VP) occurs
- As a result:
  - Drywell Average Air Temperature rises and STABILIZES at 145.6°F
  - Drywell-to-Primary Containment d/p rises and STABILIZES at +1.2 psid

Which ONE of the following describes:

(1) the required action,
and
(2) the POTENTIAL consequence of NOT taking that action?

A. (1) Restore the Drywell-to-Primary Containment d/p to within its Tech Spec limits within 1 hour.
(2) Weir wall overflow, should an inadvertent upper pool dump occur.

B. (1) Restore the Drywell-to-Primary Containment d/p to within its Tech Spec limits within 1 hour.
(2) DIRECT communication of the blowdown energy contained in the drywell airspace, to the suppression pool inventory, should a LOCA occur.

C. (1) Restore the Drywell Average Air Temperature to within its Tech Spec limits within 8 hours.
(2) Drywell temperatures in excess of the drywell STRUCTURAL design temperature, should a LOCA occur.

D. (1) Restore the Drywell Average Air Temperature to within its Tech Spec limits within 8 hours.
(2) Drywell temperatures in excess of the drywell EQUIPMENT QUALIFICATION temperatures, should a COMPLETE loss of VP occur.

Answer: B

Explanation:
B is correct – Per Tech Spec 3.6.5.4, 1.2 psid is beyond the upper limit of 1.0 psid. *Condition A requires that d/p be restored to within the limits within 1 hour.* Refer to Basis for this LCO, B 3.6.5.4, page B 3.6 – 122, the 'Background' discussion portion that reads…"The limitation on positive…". This discussion means that too high a drywell-to-CNMT can cause the vents to be already uncovered ('cleared') at the onset of a DBA LOCA (as a result of the downward force on the annulus water level). If a LOCA, then, were to occur, the RPV blowdown energy would communicate directly into the suppression pool inventory. See LP85223, Figure 2 for an illustration of this physical arrangement.

Question #	81

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	2	295010 AA2.02	55.43(b)(2)	3.9	Higher

System/Evolution Name:	Category:
High Drywell Pressure	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell pressure

A is incorrect – Part (1) is correct, but Part (2) describes the consequence of too low a d/p (i.e., below the lower LCO limit of –0.2 psid). Refer to the same page B 3.6 – 122 discussion.

C and D are incorrect – The 'stabilized' drywell average air temperature of 145.6°F is lower than the entry point for Tech Spec 3.6.5.5 (i.e., 146.53°F).

Objective:	Question Source:	Level of Difficulty:
LP85223.1.16	New	3.3

References provided to examinee:	None
References:	CPS Tech Spec 3.6.5.4, Drywell Pressure (and its Bases) CPS Tech Spec 3.6.5.5, Drywell Average Air Temperature (and its Bases) LP85223, Primary Containment

Date Written:	03/31/05	Author:	Ryder

Comments:

Although Part (1) is arguably a requirement for both RO/SRO Candidates, Part (2) is not. Part (2) asks for the potential 'consequence' of not restoring the LCO limits, which is only found in the Tech Spec Bases (as well as in the USAR). What's more, it is the Part (2) requirement that applies the KA statement's 'ability to interpret' portion. This question is in fact presented at an SRO-only level.

Question #	82

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	2	295011 2.1.14	55.43(b)(5)	3.3	Lower

System/Evolution Name:	Category:
High Containment Temperature	Emergency and Abnormal Plant Evolutions
KA Statement:	
Knowledge of system status criteria which require the notification of plant personnel	

Using the provided references, answer the following.

With the plant in MODE 3, which ONE of the following, BY ITSELF, requires NOTIFICATION of the Emergency Response Organization (ERO)?

A. Water level in the LPCS Pump Room rises to 3 inches.

B. Suppression Pool Temperature rises to 112°F.

C. Containment Temperature rises to 188°F.

D. Radiation level in RHR Pump Room 'A' rises to 10 times normal.

Answer: C

Explanation:

C is correct – Per Clinton Radiological Annex EAL's, page CL 3-8, Fission Product Barrier Matrix #1 (Containment), and the FU1 action level. Containment temperature at or above 185°F is a 'Potential Loss Containment', requiring declaration of Unusual Event. Per EP-AA-112-100-F-01, Section 1.1.D, this EAL requires NOTIFICATION of ERO personnel (station Management, only).

A is incorrect – Per EAL page CL 3-13, EOP-8 Table W, and the HA4 action level. The given water level is not above the max safe value for that room (i.e., 4 inches). Until it is, no E-Plan entry is required. The threshold for the parameter is at the ALERT action level, rather than at the UE level.

B is incorrect – Per EAL page CL 3-8, Fission Product Barrier Matrix #3 (RCS). A Suppression Pool Temperature above 110°F does <u>not</u>, by itself, require an E-Plan entry. It would, if it were coincident with a stuck-open SRV.

D is incorrect – Per EAL page CL 3-6, action level RU3. The threshold for this parameter is 1,000 times normal. An E-Plan entry is not yet required. Even if we were to consider the 'RA3' (Max Safe = 25 R/hr) threshold of Table R4. What the Candidate is expected to recognize is that the only way that a '10 times normal' level could be synonymous with having reached 25 R/hr, would be for the 'normal' radiation level to 2.5 R/hr. This conclusion would be implausible for any RHR Pump room.

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	EP-AA-1003, Clinton Radiological Annex, pages CL 3-6 thru 3-13 (<u>only!</u>)
References:	EP-AA-1003, Clinton Radiological Annex EP-AA-112-100-F-01, Site Emergency Director Checklist

Date Written:	04/01/05	Author:	Ryder
Comments: None			

Question #	83

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	209002 2.4.30	55.43(b)(1)	3.6	Higher

System/Evolution Name:	Category:
High Pressure Core Spray (HPCS)	Plant Systems

KA Statement:
Knowledge of which events related to system operations/status should be reported to outside agencies

Which ONE of the following requires a NOTIFICATION (phone call) to the NRC (OTHER than to the on-site Resident)?

A. With the plant at rated power, the MCPR value is determined to be 1.10.

B. Shift Manager discovers that one of the off-going control room operators exceeded the Technical Specification overtime guidelines WITHOUT a deviation being authorized.

C. With the plant at rated power, HPCS has been INOPERABLE for 14 days.

D. Shift Manager discovers that the LPCI 'A' quarterly surveillance, performed 30 days ago, was reviewed and signed off, but was INCOMPLETE.

Answer: C

Explanation:

C is correct – Per Tech Spec 3.5.1, Conditions B and D. Initiation of a plant shutdown (to be in Mode 3 within 12 hours) is required. Per Exelon procedure, LS-AA-1020, page 4, item 'F-aa', a 4-hour report is required for the 'initiation of a plant shutdown required by Tech Specs' (10CFR50.72(b)(2)(i)).

A is incorrect – Per Tech Spec SL 2.1.1.2. MCPR must be at or above 1.09 (2-loop). Stem conditions indicate plant is at rated power (i.e., can only be in 2-loop). No SL has been violated.

B is incorrect – This choice refers to Tech Spec (Administrative Controls) 5.2.2.e. Per LS-AA-1020, page 9, item 'T-03', because this choice describes a Tech Spec violation that solely 'administrative in nature', no NRC reporting is required. 10CFR50.72(a)(2)(i)(B) agrees with this.

D is incorrect – This choice suggests an application of Tech Spec SR 3.0.3. is in order for a now 'overdue' (beyond 25% grace period) surveillance. Operators have 24 hours to complete this surveillance and resolve this problem before having to enter any LCO. Meanwhile, no NRC notification is required, given this discovery alone.

Objective:	Question Source:	Level of Difficulty:
None	New	2.0

References provided to examinee:	None
References:	CPS Tech Spec SL 2.1.1, Reactor Core Safety Limits CPS Tech Spec 3.5.1, ECCS – Operating CPS Tech Spec 5.2.2, Unit Staff LS-AA-1020, Reportability Reference Manual 10CFR50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors

Date Written:	04/02/05	Author:	Ryder
Comments: None			

Question #	84

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	211000 A2.04	55.43(a)	3.4	Higher
System/Evolution Name:			**Category:**			
Standby Liquid Control System			Plant Systems			
KA Statement:						
Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow						

Using the provided references, answer the following.

The plant is in MODE 1, with the following:

- The Standby Liquid Control System Operability surveillance, CPS 9015.01, has just been completed
- SLC Pump 'A' flow rate ACTUAL VALUE is 41.3 gpm
- SLC Pump 'A' Dp ACTUAL VALUE is 1260 psid

Which ONE of the following:

(1) CORRECTLY INTERPRETS these surveillance results,
and
(2) describes the required action?

A. (1) SLC Pump 'A' Discharge Check Valve, 1C41-F033A, is NOT opening FULLY.
(2) Take action to establish a 6-week test frequency for 1C41-F033A.

B. (1) A blockage exists somewhere in the SLC Pump 'A' DISCHARGE.
(2) Take action to establish a 6-week test frequency for SLC Pump 'A'.

C. (1) SLC Pump 'A' Discharge Check Valve, 1C41-F033A, is NOT opening FULLY.
(2) Enter Tech Spec 3.1.7 for SLC subsystem 'A'.

D. (1) A blockage exists somewhere in the SLC Pump 'A' SUCTION.
(2) Enter Tech Spec 3.1.7 for SLC subsystem 'A'.

Answer: B

Explanation:
B is correct – Per CPS 9015.01D001, page 3, the SLC Pump flow rate (Qr) is in the ALERT Range, while the pump D/P (Dp) is much higher than normal (in fact, high outside of the Acceptable Range). Only a blockage somewhere in the pump discharge piping can yield this combination of low flow-high discharge pressure (and therefore, high d/p). Per CPS 9015.01, Section 9.1.4, personnel are directed to double the test frequency (from quarterly, to 6 weeks) when the pump goes into the ALERT Range. A and C are incorrect – Per CPS 9015.01D001, page 3, so long as the pump flow rate is at least 41.2 gpm, the discharge

Question #	84

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	211000 A2.04	55.43(a)	3.4	Higher

System/Evolution Name:	Category:
Standby Liquid Control System	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow

check valve is expected to exercise (open), fully. There is no reason to interpret that a failing check valve is in any way responsible for the low flow-high pressure combination.

D is incorrect – Although a blockage somewhere in the pump SUCTION piping may in fact yield this low flow-high pressure combination, there is no reason to declare the SLC subsystem inoperable. That is, per CPS 9015.01D001, page 3, although the 1260 psid actual value is high outside the Acceptable Range for Dp, there is no requirement to enter the Tech Spec.

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	CPS 9015.01, in its entirety CPS 9015.01D001, in its entirety
References:	CPS 9015.01, Standby Liquid Control System Operability CPS 9015.01D001, SLC Pump & Valve Operability Data Sheet

Date Written:	04/02/05	Author:	Ryder
Comments: None			

Question #	85

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	215005 2.2.25	55.43(b)(2)	3.7	Lower
System/Evolution Name:			**Category:**			
Average Power Range Monitor/Local Power Range Monitor			Plant Systems			
KA Statement:						
Knowledge of the bases in technical specifications for limiting conditions for operations and safety limits						

Per Technical Specification (or ORM) Bases, which ONE of the following identifies an APRM related Instrumentation Function that IS SPECIFICALLY relied upon by an accident analysis?

Average Power Range Monitor…

A. INOP Trip

B. INOP Rod Block

C. Neutron Flux – High, Setdown

D. Fixed Neutron Flux – High

Answer: D

Explanation:
D is correct – Per CPS Tech Spec Bases, pages B 3.3-9 and B 3.3-30a. This Function is relied upon by the Control Rod Drop Accident analysis of USAR Section 15.4.9. A and B are incorrect – Per CPS Tech Spec Bases, page B 3.3-6. This RPS Trip Function is not assumed in any safety/accident analysis; rather, it is retained in Tech Specs be virtue of being part of the NRC-approved licensing basis. The INOP Rod Block is found in the Operating Requirements Manual (ORM), where Bases Section 5.2.1 refers back to the Control Rod Block Instrumentation Bases (of Tech Spec 3.3.2.1) and Power Distribution Limits Bases (of Tech Spec 3.2). A review of these shows NO connection to any analysis that takes credit for the APRM INOP – Rod Block Function. C is incorrect – Per CPS Tech Spec Bases, pages B 3.3-6 and 7. Although this Function 'indirectly' protects Safety Limit (SL) 2.1.1.1, there is no analysis that takes direct/specific credit for this Function.

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	None
References:	CPS Tech Spec Bases B 3.3.1.1, RPS Instrumentation CPS Tech Spec Bases B 2.0, Safety Limits CPS Tech Spec Bases B 3.3.2.1, Control Rod Block Instrumentation CPS ORM, Section 2.2.1, APRM Control Rod Block Instrumentation CPS USAR, Section 15.4.9, Control Rod Drop Accident (CRDA)

Date Written:	04/04/05	**Author:**	Ryder
Comments: None			

Question #	86

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	400000 A2.03	55.41(b)(10)	3.0	Higher
System/Evolution Name:				**Category:**		
Component Cooling Water System (CCWS)				Plant Systems		
KA Statement:						
Ability to (a) predict the impacts of the following on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High/low CCW temperature						

The plant is operating at rated power, during prolonged hot summer conditions, with the following:

- Abnormally high CCW heat load conditions exist
- ALL available CCW Pumps and HXs are in service
- CCW HX Shell Side Outlet Temperature is now 106°F and STABLE

Which ONE of the following describes:

(1) a consequence of allowing CCW HX Shell Side Outlet Temperature to remain at this temperature,
<u>and</u>
(2) an appropriate action?

A. (1) Operating the CCW HX is excess of its DESIGN limit for Shell Side Outlet Temperature.
(2) Line up an FC Heat Exchanger with cooling supplied by SX.

B. (1) Operating with a CCW HX Shell Side Outlet Temperature TOO NEAR the temperature that will cause CCW Demineralizer resin damage.
(2) Bypass the CCW Demineralizer.

C. (1) UNACCEPTABLY high CCW supply temperature at the RWCU Pump seals.
(2) Secure all non-essential CCW heat loads.

D. (1) RISING radiation levels in the Fuel Building.
(2) Line up an FC Heat Exchanger with cooling supplied by SX.

Answer: A

Explanation:
A is correct – Per CPS 3203.01, Sections 6.1 and 8.3.1. The given Outlet Temperature is above the 105°F Design limit for the HXs. Section 8.3.1 directs operators to place a second FC HX in service, cooled by SX (Shutdown Service Water). Section 8.3.1.4(2)c directs operators to 'consider' shifting all FC cooling over to SX (taking about 30% of the total CCW heat load off the CCW system, per Appendix B on page 70). Close scrutiny by the CPS Facility Author and an SRO Validator (incumbent) has determined that the 'intent' of the wording in CPS 3203.01, Section 6.1.2 is as phrased here in this answer choice.   B is incorrect – Per CPS 3203.01, Section 4.6, CCW demin resin damage isn't a concern until CCW temperature nears 150°F.

Question #	86

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	400000 A2.03	55.41(b)(10)	3.0	Higher

System/Evolution Name:	Category:
Component Cooling Water System (CCWS)	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High/low CCW temperature

C is incorrect – There is no CCW supply temperature that is defined as 'unacceptably high' for the RWCU Pump Per CPS 3303.01, Section 8.1.1.11, operators must only ensure that seal cavity temperature is maintained below 200°F.

D is incorrect – There is no reference, either in CPS 3203.01 or CPS 3317.01, that associates this CCW Outlet Temperature with degraded Spent Fuel Storage Pool cooling (FC) that result in elevated radiation levels in that area.

Objective:	Question Source:	Level of Difficulty:
LP85208.1.14	New	3.0

References provided to examinee:	None
References:	CPS 3203.01, Component Cooling Water CPS 3303.01, RWCU CPS 3317.01, Fuel Pool Cooling and Cleanup CPS 5040-1C, High Temp CCW HX Outlet Temperature (alarm procedure)

Date Written:	05/16/05	Author:	Ryder

Comments:

The following justify this being an 'SRO only' question:

1. Although Part (1) of the question would apply to an RO exam, as well, in that 'knowledge' of the CCW HX design temperature limit has the procedure's Precautions/Limitation section as its source, Part (2) goes beyond this source, into the procedure's 'Abnormal' section.

2. 'Abnormals' section 8.3.1, specifically, step (2)c suggests "consideration…". Such direction is always reserved for the SRO, only. The fact that such procedural guidance exists only in an 'Operating Procedure' (i.e., CPS has no 'Loss of CCW' off-normal operating procedure) does not automatically disqualify this guidance; it is still unique to the job of the on-shift SRO.

3. Although the alarm response procedure (CPS 5040-1C), usually considered to be the RO's first procedural line of defense, includes the 'Operator Actions' to verify a properly operating temperature control valve, and to vent the CCW HXs, it does not address all of the actions found in Section 8.3.1 of the operating procedure.

Question #	87

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	202001 A2.04	55.43(b)(5)	3.7	Higher
System/Evolution Name:			**Category:**			
Recirculation System			Plant Systems			
KA Statement:						
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Single recirculation pump trip						

A plant power ascension is in progress, per CPS 3004.01, Turbine Startup and Generator Synchronization, with the following:

- Reactor Recirc Pump (RRP) 'A' is about to be transferred to FAST speed
- IMMEDIATELY BEFORE the operator positions FCV 'A' for the transfer, the following occurs:
    - RRP 'B' trips from SLOW speed to OFF
    - Operators immediately shut the RRP 'B' Discharge Valve

Which ONE of the following:

(1) PREDICTS the resulting TOTAL CORE FLOW, after flow stabilizes and BEFORE any additional operator action is taken,
<u>and</u>
(2) describes the NEXT required action?

A. (1) About 25 mlbm/hr on the 65% FCL
(2) Immediately scram the reactor, even if NO power oscillations are observed.

B. (1) About 20 mlbm/hr on the 60% FCL
(2) Verify MFLCPR is at or below 0.970.

C. (1) About 25 mlbm/hr on the 55% FCL
(2) Isolate RR Loop 'B' using CPS 3302.01, Reactor Recirculation.

D. (1) About 20 mlbm/hr on the 50% FCL
(2) Direct IMD to change the APRM setpoints to those for Single-Loop Operations.

Answer: D

Question #	87

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	202001 A2.04	55.43(b)(5)	3.7	Higher

System/Evolution Name:	Category:
Recirculation System	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Single recirculation pump trip

Explanation:

D is correct –

Part (1): Refer to CPS 3004.01, Sections 8.4.5 and 8.4.6, and to CPS 3302.01, Sections 8.1.1 and 8.1.2. The following are the initial conditions BEFORE the trip of RRP 'B': 1) Both RRPs are in SLOW speed, with their FCVs about 90% open; 2) Reactor power is about 30% with the rod line (FCL) at or near 50% FCL; 3) Thus, a review of the P/F Map indicates that Total Core Flow before the RRP 'B' trip is about 38.0 mlbm/hr. The following are the stable flow conditions AFTER the trip of RRP 'B' and the immediate shutting of its discharge valve (Immediate Operator Action, per CPS 4008.01, Section 3.2: 1) RRP 'A' remains unaffected, running in SLOW speed, with its FCV still at about 90% open; 2) Total Core Flow simply migrates down the same 50% FCL (rod line) and stabilizes at about a little more than half of the initial 38.0 mlbm/hr value (i.e., about 20 mlbm/hr). NOTE: This is the Generic Fundamental (Pumps and Fluid Flow) of losing one of two, identical, parallel-configured pumps. NOTE, ALSO: A review of the P/F Map in this area shows that there is no need to consider the precise slope of the 50% FCL (i.e., a so-called 'flatter' sloped FCL, due to fuel design). Whatever slope perturbation there might be in this area of the Map, we are still well below any point where we would expect to drift into the CONTROLLED ENTRY Region.

Part (2): A review of the CPS 4008.01 Subsequent Actions shows there is no specific action that is required 'early' in the case of this specific scenario. From Section 4.4, operators are directed to Section 4.9 for SLO. Stem conditions indicate that Sections 4.9.1, 4.9.2, and 4.9.3 are non-issues for this scenario. In Section 4.9.4, operators proceed to CPS 3005.01, to implement SLO. There, in Section 8.4.3, operators are directed to change the APRM setpoints. This is the 'next' action that is required given these stem conditions.

A is incorrect – This choice is distracting to the SRO Candidate who cannot recall (from memory, no reference is provided here) where in the plant power ascension (at what reactor power) we transfer the RRPs from Slow to Fast. If that Candidate incorrectly concludes that the transfer takes place at a power level closer to 45-50%, with a 65% rod line (FCL), then the pre-trip total core flow is about 51 mlbm/hr, and the post-trip flow is about 25 mlbm/hr (i.e., a little more than half of the initial flow). Once this is determined, a migration down the 65% FCL, places the plant either firmly in, or too close to, the Restricted Zone (scram required). The Candidate is expected to recognize that, for these reasons, a pump up-shift would not take place on the 65% FCL.

B is incorrect – This choice is distracting to the SRO Candidate who chooses the 60% FCL, with a power closer to 35%, and a pre-trip flow of about 38.0 mlbm/hr. This choice is very attractive when the SRO Candidate considers Part (2). There is clear indication, in CPS 4008.01, Section 4.2, that some priority should be given the verifying MFLCPR is at or below 0.970 for SLO. However, the suggestion, in Part (1), that the RRP up-shift will have been started from the 60% FCL makes this choice absolutely wrong.

C is incorrect – Although a 55% FCL is feasible for pump up-shift conditions, the resulting total core flow is higher than predicted (see explanation for correct answer, 'D'). In Part (2), there is NO requirement to isolate the tripped pump loop. In fact, to do so would prohibit the restoration of that loop until reactor water temperature is < 200°F (see CPS 3302.01, Section 6.11.1).

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

Question #	87

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	202001 A2.04	55.43(b)(5)	3.7	Higher

System/Evolution Name:	Category:
Recirculation System	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Single recirculation pump trip

References provided to examinee:	None
References:	CPS 3004.01, Turbine Startup and Generator Synchronization CPS 3005.01, Unit Power Changes CPS 4008.01, Abnormal Reactor Coolant Flow CPS 3302.01, Reactor Recirculation (RR)

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	88

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	219000 2.4.6	55.43(b)(5)	4.0	Higher
System/Evolution Name:				**Category:**		
RHR/LPCI: Torus/Pool Cooling Mode				Plant Systems		
KA Statement:						
Knowledge of symptom based EOP mitigation strategies						

Using the provided references, answer the following.

From rated power, operators have just manually scrammed the reactor, THEN the following occurs:

- Reactor Power is now 20%
- SLC Pumps are running
- Reactor water level is +35 inches
- Reactor Pressure is 950 psig and STABLE
- Suppression Pool Level is 19 feet and RISING SLOWLY
- One SRV has stuck open
- Suppression Pool Temperature is 93°F and RISING

From among the following actions, which ONE is required NEXT?

A. Terminate and prevent injection to establish water LEVEL BAND 'C'.

B. Drain Suppression Pool inventory to stay below the SRV Tail Pipe Limit.

C. Open a Turbine Bypass Valve to stay below the Heat Capacity Limit.

D. Place all available Suppression Pool Cooling in service.

Answer: D

Explanation:
D is correct – Per CPS EOP-6, Pool Temperature leg, top-most IF-THEN step. Pool temperature is very near point we the CRS would proceed to step 19 of this leg, where operators are directed to place all available suppression pool cooling in service. Of all the current plant conditions in the stem, this is the highest priority, currently. A is incorrect – Per EOP-1A, Level leg, and Figure G. Only through implementing the IF-AND-AND-AND-THEN override is Level Band 'C' required. Stem conditions do not support the 'Suppression Pool temperature above Figure G' portion of this override (i.e., 110°F pool temperature). B is incorrect – Per EOP-6, Pool Level leg, step 20, and Figure Q. Even though the stuck-open SRV is slowly adding to pool inventory, with level currently at 19 feet, and reactor pressure at 950 psig, we are still far below the Figure Q limit of ~23 feet. This is <u>not</u> the NEXT required action. C is incorrect – Per EOP-6, Pool Temperature leg, bottom-most step. Although intentionally lowering pressure to stay below the Heat Capacity Limit (Figure P) is allowed, even during an ATWS, the current stem conditions of 950 psig and 93°F pool temperature, are still far below the HCL limit of ~145°F pool temperature. This is not the NEXT required action.

Question #	88

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	219000 2.4.6	55.43(b)(5)	4.0	Higher

System/Evolution Name:	Category:
RHR/LPCI: Torus/Pool Cooling Mode	Plant Systems

KA Statement:
Knowledge of symptom based EOP mitigation strategies

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	EOP flowcharts
References:	CPS EOP-1, ATWS RPV Control CPS EOP-6, Primary Containment Control

Date Written:	04/07/05	Author:	Ryder
Comments: None			

Question #	89

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	2	234000 2.1.32	55.43(b)(7)	3.8	Lower
System/Evolution Name:				**Category:**		
Fuel Handling Equipment				Plant Systems		
KA Statement:						
Ability to explain and apply system limits and precautions						

Core Alterations are in progress, with the following:

- A fuel bundle has been removed from the Upper Containment Fuel Storage Pool (UCP)
- That same fuel bundle has just been placed in the core, BUT the grapple has NOT been released
- THEN, the Refuel SRO recognizes the fuel bundle is NOT in its correct core location

Which ONE of the following describes the NEXT required action?

A. RELEASE the grapple and contact the Reactor Engineer for further guidance.

B. Remove the bundle from the core and return it to its CORRECT core location.

C. Remove the bundle from the core and return it to its UCP rack location.

D. Do NOT release the grapple and contact the Reactor Engineer for further guidance.

Answer: B

Explanation:
B is correct – Refer to CPS 3703.01, Section 6.24.1. This question proposes that exact scenario. A is incorrect – This choice suggests a scenario as described in Section 6.24.2. It is <u>not</u>. C and D are incorrect – They both have strong face validity plausibility for this <u>closed</u>-reference question.

Objective:	Question Source:	Level of Difficulty:
None	New	2.0

References provided to examinee:	None
References:	CPS 3703.01, Core Alterations

Date Written:	05/03/05	**Author:**	Ryder
Comments: None			

Question #	90

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.1.4	55.43(a)	3.4	Higher

System/Evolution Name:	Category:
	Conduct of Operations

KA Statement:
Knowledge of shift staffing requirements

Considering the MINIMUM staffing requirement for either the SROs, or Fire Brigade Members, which ONE of the following describes a situation that SATISFIES the respective requirement?

A. While in MODE 1, you have a total of TWO SROs, one of whom is the Shift Manager, and the other is BOTH the Control Room Supervisor AND designated STA.

B. While in MODE 3, you have a total of FOUR Fire Brigade Members, one of whom is also the designated Safe Shutdown Qualified Operator.

C. While in MODE 5, you have a total of THREE Fire Brigade Members, ALL of whom are 'C Area Qualified'.

D. While in MODE 4, you have total of TWO SROs, NEITHER of whom is qualified as STA.

Answer: D

Explanation:

D is correct – Per OP-CL-101-102-1001, all. In Modes 4 and 5, only 2 SROs are required (for ERO functions of CRS and SM), and <u>no</u> STA is required.

A is incorrect – Per OP-CL-101-102-1001, all. In Modes 1, 2, and 3, if either the CRS or SM is also the STA, then a 3rd SRO is required and is designated as the Incident Assessor.

B is incorrect – Per OP-CL-101-102-1001, all. In ALL Modes, at least 4 Fire Brigade <u>Members</u> (plus the Leader) are required, <u>none</u> of whom can be the designated SSQO.

C is incorrect – Per OP-CL-101-102-1001, all. In ALL Modes, at least 4 Fire Brigade <u>Members</u> (plus the Leader) are required.

Objective:	Question Source:	Level of Difficulty:
None	New	2.0

References provided to examinee:	None
References:	OP-CL-101-102-1001, CPS Minimum On-Shift Staffing Functions

Date Written:	04/08/05	Author:	Ryder
Comments: None			

Question #	91

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.2.23	55.43(a)	3.8	Lower
System/Evolution Name:			**Category:**			
			Equipment Control			
KA Statement:						
Ability to track limiting conditions for operations						

IMD is about to commence a surveillance test, with the following:

- The surveillance test will cause a TECH SPEC-REQUIRED plant instrument to be INOPERABLE for the duration of the test
- Performance of the surveillance test does NOT require an LCO ACTION entry

Which ONE of the following describes a CRS required action, PRIOR to IMD beginning the surveillance test?

A. Direct the RO to hang an Adverse Condition Monitoring Tag on the annunciator window associated with the instrument.

B. Direct IMD to hang an Equipment Status Tag (EST) on the instrument, and the RO to hang a Miniature EST in the control room.

C. Identify the Technical Specification required action in the event the instrument is still INOPERABLE when the Short Duration Time Clock (SDTC) expires.

D. Identify the Maximum Out of Service Time (MOST) for the instrument and direct IMD to notify the control room if the test is still in progress within 30 minutes of the MOST.

Answer: C

Explanation:

C is correct – Per OP-AA-108-104, Sections 3.5 and 4.7.1.2, and Attachment 1. The CRS identifies the Tech Spec action that I&C will be directed to take should the instrument not be OPERABLE at the end of the Short Duration Time Clock.

A is incorrect – For an ILT Candidate, this choice has enough face validity to make it plausible, especially considering this is a closed-reference question. There is no such tag, although there is an Adverse Condition Monitoring and Contingency Planning program defined by OP-AA-108-111. That program has no connection to the situation described in the stem.

B is incorrect – The EST (and Miniature EST) is a process governed by OP-AA-108-101, and is used to track the status of equipment out of 'normal' (position). This inoperable instrument does not constitute such a condition.

D is incorrect – The MOST concept is governed by OP-CL-101-302-1001 and is bounded by equipment conditions that result in a Tech Spec entry. There is no Tech Spec entry associated with this surveillance test.

Objective:	Question Source:	Level of Difficulty:
None	New	2.3

Question #	91

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.2.23	55.43(a)	3.8	Lower

System/Evolution Name:	Category:
	Equipment Control

KA Statement:
Ability to track limiting conditions for operations

References provided to examinee:	None
References:	OP-AA-108-104, Technical Specification *Compliance* OP-AA-108-101, Control of Equipment and System Status OP-CL-101-302-1001, *ITS LCO/ORM* OR/ODCM OR Evaluations

Date Written:	04/29/05	**Author:**	Ryder
Comments: None			

Question #	92

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.2.29	55.43(b)(7)	3.8	Lower

System/Evolution Name:	Category:
	Equipment Control

KA Statement:
Knowledge of SRO fuel handling responsibilities

With all fuel handling equipment operating normally, which ONE of the following REQUIRES the DIRECT SUPERVISION of a Refuel SRO?

Transfer of...

A. NEW fuel from the Fuel Building to the Containment Refuel Floor.

B. IRRADIATED fuel from the Fuel Building to the Containment Refuel Floor.

C. NEW fuel from the New Fuel Storage Vault to the Fuel Building Transfer Pool.

D. IRRADIATED fuel from the Spent Fuel Storage Pool to the Fuel Building Transfer Pool.

Answer: B

Explanation:

B is correct – Per CPS 3703.02, Section 3.5, 1st bullet. Refuel SRO must supervise any irradiated fuel movement between the Fuel Building and Containment.

A is incorrect – Per CPS 3703.02, Section 3.5, 2nd bullet. A Reactor Engineer is allowed to supervise this transfer.

C is incorrect – Per CPS 3703.02, Section 3.5, 3rd bullet. This transfer requires no supervision by a Refuel SRO, only authorization by Control Room Supervision (SM/CRS) or Work Control (WCS).

D is incorrect – Other than the movements considered by CPS 3703.02, Section 3.5 (above), only CORE ALTERATIONS requires the supervision of a Refuel SRO. This choice does not describe a Core Alteration (i.e., there is no movement of fuel that involves the reactor vessel).

Objective:	Question Source:	Level of Difficulty:
None	New	2.0

References provided to examinee:	None
References:	CPS 3703.02, Fuel Handling Platform Operations

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	93

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.3.2	55.43(b)(5)	2.9	Higher
System/Evolution Name:				**Category:**		
				Radiological Control		
KA Statement:						
Knowledge of the facility ALARA program						

Using the provided references, answer the following.

The plant is operating at rated power, with the following:

- An operator needs to enter a Locked High Radiation Area (LHRA) to verify a valve position
- The LAST KNOWN Dose Rate (DDE at 30 cm), AT RATED POWER, was 1,200 mrem/hr for this LHRA
- The need to enter this LHRA does NOT involve any emergency situation

Which ONE of the following describes the MINIMUM radiological control REQUIREMENTS applicable to the operator's entry into this LHRA?

Can enter…

A. ONLY IF accompanied by an RP Tech; an approved RWP is NOT required.

B. ALONE; however, an approved RWP IS REQUIRED, a current survey map is NOT required.

C. ALONE; however, BOTH an approved RWP AND a current survey map ARE REQUIRED.

D. ONLY IF accompanied by an RP Tech; an approved RWP IS REQUIRED.

Answer: C

Explanation:

C is correct – Per RP-AA-460, Section 4.7.1, an RP Tech can substitute for a current survey map. Per Section 4.4, RP procedures do require an approved RWP for HRA and LHRA entries (other than for emergent entries).

A is incorrect – Per RP-AA-460, Section 4.4 (as described above), RP procedures require the RWP. However, this choice is very plausible to the Candidate who opts for applying the exemption of Tech Spec 5.7.4, without regard for the fact that to enter without the RWP would amount to NOT operating in accordance with 'plant radiation protection procedures…'

B is incorrect – Per RP-AA-460, Section 4.7.1, accompaniment by an RP Tech is required if a current survey map is not available.

D is incorrect – For the reasons associated with the correct answer, 'C'.

Question #	93

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.3.2	55.43(b)(5)	2.9	Higher

System/Evolution Name:	Category:
	Radiological Control

KA Statement:
Knowledge of the facility ALARA program

Objective:	Question Source:	Level of Difficulty:
None	New	3.0

References provided to examinee:	CPS Technical Specification 5.7, in its entirety (2 pages) RP-AA-460, in its entirety
References:	CPS Technical Specification 5.7, High Radiation Area RP-AA-460, Controls for High and Very High Radiation Areas

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	94

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.3.1	55.43(b)(4)	3.0	Lower
System/Evolution Name:			**Category:**			
			Radiological Control			
KA Statement:						
Knowledge of 10 CFR 20 and related facility radiation control requirements						

Consider the following related to one of your crew's NLOs:

- After discovering she is ALREADY 3 MONTHS PREGNANT, she formally submits a 'Declaration of Pregnancy', TODAY
- Exposure records reveal that she has received 100 mrem (DDE) in the LAST 3 MONTHS
- She IS choosing to abide by the work restrictions prescribed in a Dose Equivalent Reduction Action Plan

Which ONE of the following identifies:

(1) when her work restrictions AUTOMATICALLY expire,
and
(2) how many ADDITIONAL mrem (DDE) she (including the embryo/fetus) is allowed to receive during the REMAINDER of her pregnancy?

A. (1) When she is no longer pregnant
(2) 400 mrem

B. (1) 12 months from today's date
(2) 400 mrem

C. (1) 12 months from today's date
(2) 500 mrem

D. (1) When she is no longer pregnant
(2) 350 mrem

Answer: B

Explanation:
B is correct – Per RP-AA-270, *Attachment 3, page 1 of 1. Given that she has received a total* DDE of only 100 mrem since becoming pregnant, she is limited to a total of 500 mrem DDE for the entire pregnancy, or an additional 400 mrem from the remainder of her pregnancy. Unless she withdraws her declaration before-hand, this declaration and its work restrictions automatically expire 12 months from today. A, C, and D are incorrect – Each is a plausible mis-understanding, or mis-application, of the Attachment 3 requirements cited above.

Objective:	Question Source:	Level of Difficulty:
None	New	4.0

Question #	94

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.3.1	55.43(b)(4)	3.0	Lower

System/Evolution Name:	Category:
	Radiological Control

KA Statement:
Knowledge of 10 CFR 20 and related facility radiation control requirements

References provided to examinee:	None
References:	RP-AA-270, Prenatal Radiation Exposure

Date Written:	04/11 /05	Author:	Ryder

Comments:

This question is categorized as Lower Cognitive (LCL) because it only requires the recall of two, independent pieces of information: 1) 500 mrem for the entire pregnancy, and 2) 12 months for the automatic expiration of the declaration's work restrictions. There is no IF-THEN relationship that necessarily exists between the two parts of the question.

This is an **SRO-only** question because the 'Declaration' is one that a 'Work Supervisor' must review and approve. In fact, the Work Supervisor (CRS/SM in her case), is critical in stipulating the work restrictions for the Dose Equivalent Reduction Action Plan.

Question #	95

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.1.34	55.43(b)(5)	2.9	Higher
System/Evolution Name:				**Category:**		
				Conduct of Operations		
KA Statement:						
Ability to maintain primary and secondary plant chemistry within allowable limits						

Using the provided references, answer the following.

FOLLOWING a refueling outage, the plant entered MODE 2 in preparation for a plant startup 8 hours ago, with the following:

- Reactor water temperature is 215°F
- Reactor Power is at the Point of Adding Heat (POAH)
- Chemistry makes the following sample data available to the CRS:
  - Feedwater Conductivity is 0.30 μS/cm
  - Feedwater Oxygen is 280 ppb
  - Reactor Coolant Conductivity is 0.50 μS/cm
  - Reactor Coolant Chlorides is 150 ppb
- Continuous Conductivity and Oxygen monitors AGREE with the above sample data

Which ONE of the following describes the NEXT required action?

A. Direct Chemistry to obtain and analyze a confirmation sample of REACTOR COOLANT within 8 hours.

B. Direct Chemistry to obtain and analyze a confirmation sample of FEEDWATER within 4 hours.

C. Suspend control rod withdrawals in preparation for returning to MODE 3.

D. Notify the Nuclear Operations Duty Officer of an Action Level 2 condition.

Answer: D

Explanation:
D is correct – Per CY-AB-120-100, Section 4.3.2. For these plant conditions (POAH, Mode 2), Reactor Coolant Chlorides are higher than the Action Level 2 limit (100 ppb). Per Attachment 2, the <u>first</u> requirement is to notify the Nuclear Operations Duty Officer of the Action Level 2 condition. A is incorrect – This choice has face validity (psychometrically balanced with choice 'B'), and is intended for the Candidate who incorrectly applies Note 'a' of CY-AB-120-100, Section 4.3.2.1 Table. B is incorrect – This choice is plausible to the Candidate who recognizes Feedwater Conductivity is higher than the Action Level 1 limit of CY-AB-120-110, Table 1a, and then applies the Action Level 1 decision tree of Attachment 1.

Question #	95

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.1.34	55.43(b)(5)	2.9	Higher

System/Evolution Name:	Category:
	Conduct of Operations

KA Statement:
Ability to maintain primary and secondary plant chemistry within allowable limits

However, this would be a mistake in the light of Note 'a' associated with the Table 1a limits. With the given stem conditions, the steam jet air ejectors cannot be in service (placed in service at or above 150 psig); reactor power is only *at the point of adding heat* (about IRM Range 6 or 7).

C is incorrect – One of the open-references provided to the Candidate for this question is the ORM section 2.3.1 for Reactor Coolant Chemistry. Table 3.3.1-1 shows that the given 150 ppb value for reactor coolant chlorides is above the 0.1 ppm (i.e., 100 ppb) limit for Modes 2 and 3. The Candidate may refer to OR Action 3.3.1.2, but neglect to notice that the action to return to Mode 3 within 12 hours is not required until after this chloride value has been exceeded for 48 hours.

NOTE – The Feedwater Oxygen sample value given in the stem is there for psychometric balance between Feedwater and Reactor Coolant. The stem statement regarding …'8 hours ago' is there to ensure any consideration of applying Note 'c' of CY-AB-120-100, Section 4.3.2.1 Table, is avoided.

Objective:	Question Source:	Level of Difficulty:
None	New	2.5

References provided to examinee:	CPS Operating Requirements (ORM) 2.3.1, entire section CY-AB-120-100, in its entirety CY-AB-120-110, in its entirety
References:	CPS OR 2.3.1, Reactor Coolant System Chemistry CY-AB-120-100, Reactor Water Chemistry CY-AB-120-110, Condensate and Feedwater Chemistry

Date Written:	04/29/05	Author:	Ryder
Comments: None			

Question #	96

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.4.41	55.43(b)(5)	4.1	Higher
System/Evolution Name:			**Category:**			
			Emergency Procedures and Plan			
KA Statement:						
Knowledge of the emergency action level thresholds and classifications						

Using the provided references, answer the following.

The plant is operating at rated power, when the following occurs:

- At Time = 0 minutes, ALL annunciators on P877 are lost due to a blown power supply
- At Time = +20 minutes, an UNISOLABLE primary system discharge causes operators to enter EOP-8 because an Area Temperature has JUST REACHED its EOP-8 entry value
- At Time = +55 minutes, as directed by EOP-8, operators perform an RPV Blowdown

Which ONE of the following identifies the LATEST time:

(1) by when the FIRST required State/Local agency NOTIFICATION must be completed, and
(2) by when the event MUST be ESCALATED to the HIGHEST Classification Level necessary for these plant conditions?

A. (1) Time = +30 minutes
(2) Time = +35 minutes

B. (1) Time = +45 minutes
(2) Time = +40 minutes

C. (1) Time = +50 minutes
(2) Time = +70 minutes

D. (1) Time = +85 minutes
(2) Time = +70 minutes

Answer: C

Explanation:

C is correct – Part (1): The earliest that an EAL threshold is reached is at Time = +15 minutes, for EAL 'MU6' (see CPS Annex page CL 3-11). Per EP-AA-112-100, Section 2.1, the Shift Manager (SM) would have until Time = +30 minutes to classify/declare the event as a UE, and until Time = +45 minutes to complete the required State/Local notifications. However, at Time = +20 minutes, the 'FA1' EAL threshold is reached due to a 'Potential Loss of RCS' (see Annex page CL 3-8). Again, the SM would have until Time = +35 minutes (20 + 15 = 35) to classify/declare the event as an ALERT. Per EP-AA-111, Section 4.1, the 2nd NOTE, once this higher classification level is declared, if the UE notification has not yet been made, the UE event is essentially dismissed (without further consideration), in favor of

Question #	96

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	3	Generics	2.4.41	55.43(b)(5)	4.1	Higher
System/Evolution Name:				**Category:**		
				Emergency Procedures and Plan		
KA Statement:						
Knowledge of the emergency action level thresholds and classifications						

the more 'severe' ALERT event declaration. In other words, given these stem conditions, the UE event (loss of annunciators) does not result in a 'First required' State/Local agency notification. Rather, the SM has until Time = +50 minutes to complete the ALERT notifications. And since the next plant transient that requires a re-classification (escalation) to an SAE (i.e., the RPV Blowdown) doesn't even occur until Time = +55 minutes, the SM does in fact get a chance to complete the ALERT notifications at Time = +50 minutes. This, therefore, amounts to the 'First required' State/Local agency notification for these given plant conditions. Part (2): An SAE is the highest classification required for these plant conditions (i.e., the 'FS1' EAL is reached due to Loss of Containment; see Annex page CL 3-8). Again, per EP-AA-112-100, Section 2.1, the SM must declare this escalation (from an ALERT) no later than Time = +70 minutes (+55 + 15 minutes = +70 minutes).

A is incorrect – For the reasons already described above. Part (1) is plausible to the Candidate who disregards the EP-AA-111, Section 4.1, requirements, and mistakenly applies a +30 minute requirement (+15 + 15 = +30 minute) of EP-AA-112-100, Section 2.1, to the start of the 'threshold clock' for 'MU6'. Part (2) is plausible to the Candidate who recognizes the need to escalate to an ALERT by no later than Time = +35 minutes (FA1 threshold at Time = +20, +15 minutes to classify, per EP-AA-112-100, Section 2.1). This Candidate does not recognize that the RPV Blowdown at Time = +55 minutes results in a further escalation to an SAE ('FS1' EAL).

B is incorrect – For the reasons already described above. Part (1) is plausible to the Candidate who, although correctly waits for the MU6 threshold clock to become 'active' (i.e., the threshold is met) before applying the +30 minute allowance of EP-AA-112-100, Section 2.1, fails to apply the EP-AA-111, Section 4.1 requirement that essentially dismisses the MU6 event. Part (2) is designed to provide psychometric balance with Part (2) of choice 'D' (i.e., a time value that is earlier than its associated Part (1) value). It has sufficient face validity for the thoroughly confused Candidate, as well.

D is incorrect – For the reasons already described above. This choice (both Parts) is plausible to the Candidate who cannot effectively translate the earlier of the EOP-8 actions identified in the stem conditions, and instead simply applies the final state of the plant (RPV Blowdown is progress) and concludes that EAL 'FS1' applies. This Candidate will necessarily recognize that the SM has 15 minutes to classify the SAE (i.e., Time = +55 minutes + 15 minutes = +70 minutes), yielding Part (2) of the answer choice. Similarly, the SM has an additional 15 minutes, from Time = +70 minutes, to complete the State/Local notifications (Time = +70 + 15 minutes = +85 minutes), yielding Part (1) of the answer choice.

Objective:	Question Source:	Level of Difficulty:
LP87537.1.10	New	3.3

References provided to examinee:	EP-AA-1003, Clinton Radiological Annex, pages CL 3-6 thru 3-13 EOP flowcharts
References:	EP-AA-1003, Clinton Radiological Annex EP-AA-112-100, Control Room Operations EP-AA-111, Emergency Classification and PARs CPS EOP-8, Secondary Containment Control

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	97

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295031 2.1.20	55.43(b)(5)	4.2	Higher
System/Evolution Name:			**Category:**			
Reactor Low Water Level			Plant Systems			
KA Statement:						
Ability to execute procedure steps						

Using the provided references, answer the following.

An ATWS and LOCA are in progress, with the following:

- Reactor pressure is 600 psig and slowly lowering
- Operators are injecting with ALL available PREFERRED ATWS Systems
- Reactor water level is –149 inches Wide Range and slowly lowering
- Containment Temperature is 175°F and slowly rising
- Containment Pressure is 2.0 psig and slowly rising
- Suppression Pool Level is 19 feet, 5 inches, and slowly rising

Which ONE of the following describes the NEXT required action?

A. Start Containment Sprays.

B. Leave Level and Pressure; enter EOP-2.

C. Leave Level and Pressure; enter EOP-3.

D. Implement the actions of CPS 4411.05 for rising pool level.

Answer: C

Explanation:

C is correct – Refer to EOP-1A, Level leg, and Detail C. As soon RPV level drops to –150" Wide Range, this instrument becomes unusable, with a containment temperature above 100°F (175°F is indicated in the stem conditions). Operators must immediately transition to the Fuel Zone Range instruments. Because reactor pressure is about 600 psig (still far above the 'depressurized' (0 psig) calibration conditions for the Fuel Zone instruments), and Wide Range instruments are reading essentially the same as actual level before they become unusable, the Fuel Zone will indicate well below TAF (-162") when the operators operationally transition to them. As such the CRS has no choice but to implement the bottom-most step of the EOP-1A Level leg. The NEXT action is to 'Leave Level and Pressure, and enter EOP-3 to Blow Down'.

A is incorrect – Per EOP-6, Containment Temperature leg, and Figure O. The existing Containment Temperature (175°F)/Containment Pressure (2.0 psig) combination has us on the 'bad' side of the Containment Spray Initiation Limit curve, Figure O. Until things change (likely to be an additional rise in Containment Pressure), it is <u>not</u> OK to Spray. This choice is <u>not</u> the NEXT required action.

B is incorrect – This would be the NEXT action if there were <u>no</u> usable RPV water level instruments (i.e., no available Fuel Zone instruments) when the Wide Range instruments dropped below the minimum usable level of Detail C. The CRS would invoke the top-most override step of the EOP-1A Level leg, by declaring RPV water level unknown and transitioning to EOP-2 for RPV flooding. Because there are <u>no</u> indications in the given stem conditions to cause the Candidate to conclude that the Fuel Zones are not available, this choice is <u>not</u> the NEXT required action.

Question #	97

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295031 2.1.20	55.43(b)(5)	4.2	Higher

System/Evolution Name:	**Category:**
Reactor Low Water Level	Plant Systems

KA Statement:
Ability to execute procedure steps

D is incorrect – This choice suggests the need to give a priority to the slowly rising suppression pool level (per the Pool Level leg of EOP-6). This procedure (CPS 4411.05) is associated not only with a pool level high enough to threaten the SRV Tail Pipe Limit of Figure Q, but in fact provides the actions necessary to protect in-Containment equipment in the event that such equipment becomes submerged. Stem conditions suggest a pool level that is no where near this high level.

Objective:	Question Source:	Level of Difficulty:
	New	2.0

References provided to examinee:	EOP flowcharts
References:	CPS EOP-1A, ATWS RPV Control CPS EOP-6, Primary Containment Control

Date Written:	05/03/05	**Author:**	Ryder
Comments: None			

Question #	98

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295003 AA2.04	55.43(b)(5)	3.7	Higher
System/Evolution Name:			**Category:**			
Partial or Complete Loss of A.C. Power			Emergency and Abnormal Plant Evolutions			
KA Statement:						
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: System lineups						

Using the provided references, answer the following.

With the plant operating at rated power, a COMPLETE LOSS of AC Power (including Div 3) occurred 15 MINUTES AGO, and is still in progress, with the following:

- RCIC is being used to control reactor water level at about +35 inches
- SRVs are being used to control reactor pressure between 800 and 1065 psig
- THEN, power is returned to the station via the RAT

Which ONE of the following identifies the AC buses that should be re-energized FIRST?

A. Div 1

B. Div 2

C. Div 3

D. BOP

Answer: B

Explanation:
B is correct – Refer to CPS 4200.01, Section 4.2.3 NOTE. This question requires the SRO Candidate to consider the overall existing plant status in light of the sustained ability of RCIC to maintain RPV inventory, and the impact of the SRVs adding energy (heat) to the Suppression Pool. All Divisional batteries (including the Div 1 battery for RCIC support) are 4-hour batteries (see USAR, Section 8.3.2.1.2.1). With RCIC adequately controlling level, with there being plenty of decay heat and, therefore, steam pressure to support RCIC, and with RCIC having been on its battery for only 15 minutes, now, there is <u>no</u> urgent need to restore power to the Div 1 battery charger. Before the loss of AC occurred, suppression pool level is understood to have been within Tech Spec limits (at least 19 feet, per Tech Spec 3.6.2.2), and suppression pool temperature significantly below its Tech Spec limit of 95°F, per Tech Spec 3.6.2.1. Refer to Figure P, Heat Capacity Limit, of EOP-6. With post-scram reactor pressure between 800 and 1065 psig, the pool's Heat Capacity Limit is no where near being threatened (happens as pool temperature nears ~145°F). Therefore, there is <u>no</u> urgent need to restore any of the Divisional (1, 2, 3, 4) power necessary to re-establish the main condenser as the preferred heat sink. With RCIC capably controlling level (as already described), there is <u>no</u> urgent need to restore Div 3 power to enable HPCS as an alternate injection source. With the RCIC/SRV feed/bleed combination providing adequate core cooling and there being no urgent need to restore the main condenser, or a vacuum, there is no immediate need for restoring Reactor Recirc, CCW, Plant Chilled Water, Plant Service Water, Plant Air, Condensate, or a CRD Pump. Therefore, there is <u>no</u> urgent need to restore non-divisional (BOP) bus power. In the end, the SRO Candidate should recognize that restoring Div 2 bus power is the <u>highest</u> priority, because it re-energizes the plant security systems, allowing card-reader access throughout the plant to support system restorations, and because it re-energizes 4160V Bus 1B1, which supports the main turbine turning gear (which otherwise has to be manually jacked to prevent post-shutdown rotor bowing; see CPS 4200.01, Section 4.2.1).

Question #	98

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	1	295003 AA2.04	55.43(b)(5)	3.7	Higher

System/Evolution Name:	**Category:**
Partial or Complete Loss of A.C. Power	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: System lineups

A, C, and D are incorrect – For the reasons described above.

Objective:	Question Source:	Level of Difficulty:
None	New	2.3

References provided to examinee:	CPS 4200.01, **excluding**: pages 3, 4, 14-20, and **all** of the Appendices (A, B, and C); double jeopardy concerns
References:	CPS 4200.01, Loss of AC Power USAR, Section 8.3.2.1.2.1, Batteries CPS Tech Spec 3.6.2.1, Suppression Pool Average Temperature CPS Tech Spec 3.6.2.2, Suppression Pool Water Level

Date Written:	04/13/05	**Author:**	Ryder
Comments: None			

Question #	99

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	2	500000 EA2.03	55.43(b)(5)	3.8	Higher
System/Evolution Name:				**Category:**		
High Containment Hydrogen Concentration				Emergency and Abnormal Plant Evolutions		
KA Statement:						
Ability to determine and/or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATION: Combustible limits for drywell						

Using the provided references, answer the following.

An ATWS is in progress, with the following:

- Operators are controlling RPV level using Level Band 'C'
- THEN, a LOCA occurs
- Operators enter EOP-3 and open 7 ADS Valves
- Hydrogen Igniters have tripped OFF (cause unknown) and are still OFF
- NOW, the MCR Hydrogen Monitors have been warmed up and are JUST beginning to come on-scale
- THEN, BOTH MCR Hydrogen Monitors STOP working (fail downscale, cause unknown)

Which ONE of the following describes the NEXT required action?

A. Prevent Igniter restart.

B. Attempt to re-start the Igniters.

C. Direct Chemistry to sample for hydrogen.

D. Slowly inject with Preferred Systems to re-establish Level Band 'C'.

Answer: A

Explanation:
A is correct – Refer to EOP-7, top-most override step, or to the EOP-6, top-most override step. The SRO Candidate may believe that when the hydrogen monitors stop working, the NEXT action is to 'sample…for hydrogen' per 4412.00C001. However, the Candidate is expected to recognize that a Blowdown having been directed by EOP-1A *means that RPV level dropped below TAF (see the bottom-most step of* EOP-1A Level leg). How should the SRO Candidate interpret this with respect to actual hydrogen levels, despite the lack of functional monitors? Refer to the EOP Technical Bases for EOP-7, page 9-6. This discussion proposes that the CRS should use 'judgment based on plant conditions' to determine what actual hydrogen levels must be. Here, it suggests that with RPV level below TAF, hydrogen production should be suspected; and so, with the absence of monitors, the CRS should consider that actual levels do exist, and therefore require mitigation by EOP-7 actions. Refer to CPS 4412.00C001, page 2 of 2, Section 3.1. Because of the removal of the PASS panel H2/O2 monitors at CPS, this section instructs operators to consider hydrogen and oxygen levels to be 'unknown', if both MCR Monitors stop working. In either case, the NEXT action, therefore, is to 'prevent igniter restart' per the top-most override of EOP-7. Stem conditions do indicate that the igniters are currently off. B is incorrect – For the reasons described above.

Question #	99

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	1	2	500000 EA2.03	55.43(b)(5)	3.8	Higher

System/Evolution Name:	Category:
High Containment Hydrogen Concentration	Emergency and Abnormal Plant Evolutions

KA Statement:
Ability to determine and/or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATION: Combustible limits for drywell

C is incorrect – This is the most likely choice for the uncertain Candidate. If the Candidate either, fails to consider the 'below TAF…hydrogen production…levels unknown' concept presented in the EOP Bases, or fails to recall the explicit requirements of CPS 4412.00C001, Section 3.1, he/she will likely apply the top-most IF-THEN action of the EOP-7 override. It is important to understand why this choice does not represent a 2nd correct answer…An understanding/application of the EOP-7 Bases on page 9-6 is ALL that is required for the CRS to determine that the NEXT required action is to prevent igniter restart, having declared the hydrogen levels to be 'unknown' For this choice to be correct, would mean that only by first going to the 4412.00C001 procedure could the CRS declare the hydrogen levels to be unknown…this is not true. What's more, the choice's wording suggests that the NEXT action would be to wait for Chemistry's actual sample (not readily available with the absence of the PASS monitors). In the meantime, if the igniters were to be started with a high hydrogen concentration present, a combustible situation could result. Conclusion: this choice does not represent the NEXT required action.

D is incorrect – Once 7 ADS Valves are open in EOP-3, that EOP directs operators to return to the Level leg of EOP-1A. However, upon returning to EOP-1A, at step 7, operators must wait until the RPV has depressurized below 138 psig, per Table J, before attempting to re-establish Level Band 'C'.

Objective:	Question Source:	Level of Difficulty:
None	**New**	3.0

References provided to examinee:	EOP flowcharts
References:	CPS EOP-1A, ATWS RPV Control CPS EOP-3, RPV Blowdown CPS EOP-6, Primary Containment Control CPS EOP-7, Hydrogen Control CPS 4412.00C001, Sampling Containment and Drywell for Hydrogen CPS EOP Technical Basis document

Date Written:	05/16/05	Author:	Ryder
Comments: None			

Question #	100

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	239002 A2.03	55.43(b)(5)	4.2	Higher
System/Evolution Name:				**Category:**		
Relief/Safety Valves				Plant Systems		
KA Statement:						
Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV						

A plant pressurization is in progress, with REACTOR PRESSURE AT 750 PSIG, when the following occurs:

- ONE SRV inadvertently opens and is STUCK OPEN
- The Immediate Operator Actions for CPS 4009.01, Inadvertent Opening Safety/Relief Valve, HAVE been performed
- The SRV is still STUCK OPEN

Which ONE of the following:

(1) identifies an expected tailpipe temperature, for the STUCK OPEN SRV, on the temperature recorder at P614,
<u>and</u>
(2) describes the required operator action?

A. (1) 270°F
(2) Attempt to shut the SRV by pulling its solenoid fuses; if the SRV remains open, shift CNMT HVAC to CCP Filtered Mode, then place the Mode Switch in SHUTDOWN.

B. (1) 310°F
(2) Attempt to shut the SRV by pulling its solenoid fuses, AND enter CPS 4005.01, Loss of Feedwater Heating.

C. (1) 380°F
(2) Attempt to shut the SRV by pulling its solenoid fuses; if the SRV remains open, shift CNMT HVAC to CCP Filtered Mode, then place the Mode Switch in SHUTDOWN.

D. (1) 400°F
(2) Attempt to shut the SRV by pulling its solenoid fuses, AND enter CPS 4005.01, Loss of Feedwater Heating.

Answer: C

Explanation:
C is correct – Per CPS 4009.01, Section 1.1, tailpipe temperature is expected to be >375°F for a stuck open SRV (Note: *this value is derived from empirical CPS test data, and is recognized as being <u>inconsistent</u> with the predicted* temperature derived from the Steam Tables/Mollier Diagram.). Per Section 4.3, operators will attempt to shut the SRV

Question #	100

RO/SRO:	Tier:	Group:	KA:	CFR	SRO IR:	Cog Level
SRO	2	1	239002 A2.03	55.43(b)(5)	4.2	Higher

System/Evolution Name:	Category:
Relief/Safety Valves	Plant Systems

KA Statement:
Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV

by pulling its solenoid fuses. Per Section 4.7, if it remains open, operators are directed to shift CNMT Bldg HVAC, then perform a Rapid Plant Shutdown IAW CPS 3005.01. Per CPS 3005.01, Section 8.5.1, a Rapid Plant Shutdown looks like the following: 1) Evacuate Containment (already done as part of the Immediate Operator Action 3.1, of CPS 4009.01); 2) lower core flow to 43 mlbm/hr with Recirc (because the stem conditions indicate the plant is only in MODE 2 (pressurization in progress, at 750 psig), both RR Pumps are still running in SLOW speed and total core flow is already <43 mlbm/hr; therefore, this 'action' is already done; and 3) place the Mode Switch in SHUTDOWN; this is the only action remaining to complete a Rapid Plant Shutdown, given these stem conditions.

A is incorrect – Per CPS 4009.01, Section 1.1, this is an expected temperature for a leaking SRV, not an open SRV.

B is incorrect – Per CPS 4009.01, Section 1.1, this is the Hi-Hi Temperature alarm setpoint, but is still far below the range (>375°F) expected for an open SRV. Additionally, even though a 'stuck open SRV' is an entry condition (Symptom) for CPS 4005.01, Loss of Feedwater Heating, there is NO entry required given these stem conditions (i.e., at 750 psig, in MODE 2, the plant is well below the 21.6% reactor power applicability threshold for the Loss of Feedwater Heating off-normal (see CPS 4005.01, NOTE at the top of page 2 of 8).

D is incorrect – Even though 400°F may be an expected tailpipe temperature for a stuck open SRV (i.e., >375°F), the second part of this choice is incorrect for the same reason attributed to choice 'B'.

Objective:	Question Source:	Level of Difficulty:
DB400901.1.3.1	New	2.5

References provided to examinee:	None
References:	CPS 4005.01, Loss of Feedwater Heating CPS 4009.01, Inadvertent Opening Safety/Relief Valve CPS 3005.01, Unit Power Changes CPS 3002.01, Heatup and Pressurization

Date Written:	04/14/05	Author:	Ryder
Comments: None			