The plant is operating at 100% power.

Which of the following describes the effect of pushing the STOP pushbutton for RRC-P-1B?

A. RRC-P-1B trips. RRC-P-1A runs back to 51 hertz.

B. RCC-P-1B trips. RRC-P-1A continues to run at 60 hertz.

C. RRC-P-1B runs back to 51 hertz, RRC-P-1A continues to run at 60 hertz.

D. RRC-P-1B runs back to 30 hertz, RRC-P-1A runs back to 30 hertz.

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	202002A1.01 3.2/3.2 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the Recirculation System controls including: Recirculation Pump Flow
REFERENCE:	LO000184 rev 13, pages 15, 16, 21, and 22
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #1 RO T2, GP1, #2
LO:	9681 – Describe the operation of the following ASD controls on P602, including expected indications and system response: STOP pushbutton.
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	Pushing the STOP pushbutton causes both ASD channels to trip and open both the supply and load breakers. This causes the pump to trip. The trip has no effect on the remaining loop. B is correct.
COMMENTS:	

ex98030

The reactor was operating at 99% power when the reactor scrammed. The following conditions exist:

TR-S has tripped on overcurrent
HPCS and RCIC auto started, RCIC is maintaining level in the normal band
Drywell pressure is .92 psig
Both Recirc pumps have tripped CB-RPT-3A/4A and CB-RPT-3B/4B are open
Both feed pumps are off

Which of the following caused the scram?

A. Reactor level + 8 inches

B. MSIV isolation

C. Reactor pressure 1069 psig

D. Main turbine trip

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295005AK2.03 3.2/3.3 10CFR55.41 & 45 – Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Recirculation System.

REFERENCE: LO000196 rev. 12, pages 13 and 14

SOURCE: **BANK QUESTION – 1998 NRC EXAM -** SRO T1,G2, #4 RO T1, G1, #5

LO: 5023 – Predict the impact on the RRC System of each of the following conditions or events: e. EOC-RPT logic.

RATING: H3 H4

ATTACHMENT: NONE

JUSTIFICATION: The only trip that causes <u>BOTH</u> CB-RPT-3A (3B) and 4A (4B) to open is the EOC-RPT trip coming from a turbine trip.

COMMENTS:

A single failure has just occurred (with no other failures) causing an inboard and outboard NS4 isolation. The plant continues to operate at 99% power.

Which of the following is the cause of this isolation?

A. 86 lockout on Bkr 7-1

B. 86 lockout on Bkr 8-3

C. Loss of SL-71

D. Loss of SL-81

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295003AK3.06 3.7/3.7 10CFR55.41 & 45 – Knowledge of the reason for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Containment Isolation.

REFERENCE: LO000182 pages 56, 57, and 88 LO000161 page 16, LO000173 page 13

SOURCE: **NEW QUESTION** SRO T1, GP1, #1 RO T1, GP2, #3

LO: 5604 – List the actions that would occur on a loss of one or both RPS power supplies to the NS4 Logic.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The cause for the inboard and outboard isolation is a loss of RPS B. B is the only selection that causes a loss of RPS B. A causes a loss of RPS A, which would result in the closure of the outboard valves only. C and D would not result in any closure of isolation valves.

COMMENTS:

The plant is operating at 100% power when a jet pump nozzle separates from the rams head.

Which of the following describes expected plant indications?

RRC loop flow on the loop with the failed nozzle.....

A. increases, actual core flow decreases, indicated core flow increases, and core thermal power decreases.

B. decreases, actual core flow decreases, indicated core flow increases, and core thermal power decreases.

C. increases, actual core flow increases, indicated core flow decreases, and core thermal power increases.

D. decreases, actual core flow increases, indicated core flow increases, and core thermal power increases.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295001AK2.02 3.2/3.3 10CFR55.41 & 45 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Nuclear Boiler Instrumentation
REFERENCE:	LO000196 rev/ 12. page 29
SOURCE:	**BANK QUSTION – 98 NRC EXAM –** SRO T1, GP2, #2 RO T1, GP2, #1
LO:	5023 – Predict the impact on the RRC System of each of the following conditions or events: a. Jet Pump Failure
RATING:	H3 H4
ATTACHMENT:	NONE
JUSTIFICATION:	A is correct because when the failure occurs, RRC flow increase due to lower resistance, actual core flow decreases because there are fewer jet pumps in operation. Indicated flow increases because there is more flow through the failed jet pump, this is reverse but the flow indicators do not differentiate between forward and reverse flow. Core thermal power decreases due to the decrease in core flow.

The plant had been operating for an extended run at full power when an accident occurred. Primary Containment pressure is approaching the Primary Containment Pressure Limit (PCPL).

Which of the following is correct for these conditions?

A. The Drywell must be vented to prevent a failure of the wetwell to drywell interface.

B. The Wetwell must be vented to prevent a failure of the wetwell to drywell interface.

C. Primary containment must be vented irrespective of offsite release to prevent failure of the containment and subsequent loss of ECCS Systems.

D. Primary containment can be vented to prevent failure of the containment and subsequent loss of ECCS Systems when the projected offsite release rate is within limits.

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 295010 2.4.6 3.1/4.0 10CFR55.41, 43.5, & 45 – Knowledge of symptom base EOP mitigation strategies during High Drywell Pressure.
REFERENCE: PPM 5.0.10 rev 6, pages 93, 94, and 260
SOURCE: **NEW QUESTION –** SRO T1, GP1, #7
LO: 8364 – Given a list, identify the two reasons for venting the primary containment irrespective of offsite release rates when the Primary Containment Pressure Limit is reached.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: PPM 5.0.10 states that the containment must be vented prior to exceeding the PCPL to prevent potential containment failure and the potential loss of ECCS Systems. C is correct. D is incorrect because containment integrity is maintained at this point irrespective of offsite release. A and B are incorrect because PCPL has no effect on the WW to DW interface.
COMMENTS:

The plant was operating at 100% power when a fire caused the abandonment of the Control Room. The CRO1, at the Remote Shutdown Panel is attempting to contact the CRS at the Alternate Remote Shutdown Panel. The door between the Remote and Alternate Remote Shutdown rooms is jammed closed and cannot be opened.

Which of the following describes the communication systems permanently installed at both of these panels for communication with the other room?

A. Plant page and plant radio

B. Plant page and plant phones

C. Sound powered phones and plant radio

D. Sound powered phones and plant phones

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295016AK2.02 4.0/4.1 10CFR55.41 - Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations.

REFERENCE: LO000210 rev. 5, pages 3 & 4

SOURCE: **BANK QUESTION – 2000 NRC EXAM– Direct –** SR0 T1, GP1, #6 RO T1, GP2, #8

LO: 7739 – State the communications systems available at the Remote Shutdown room.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Sound powered phones and Plant phones are the only systems available at both the RSD Panel and ARSD Panel. D is the correct answer. The other 3 are all incorrect because one or both are not available in either the RSD or the ARSD Panels.

COMMENTS: This question has been slightly modified. It is still counted as direct from the bank.

The EOPs direct entering PPM 5.1.1 RPV Control when suppression pool level cannot be maintained greater than 19 feet 2 inches.

Which of the following is the basis for this direction?

A. Specific directions for reactor level control with low suppression pool level are given in PPM 5.1.1 RPV Control.

B. Steam condensation during a LOCA cannot be assured with suppression pool level less than 19 feet 2 inches.

C. Suppression pool volume below 19 feet 2 inches is not adequate for steam condensation during a 100% power ATWS

D. The conditions for entry into PPM 5.2.1 Primary containment Control may not have caused an entry into PPM 5.1.1 RPV Control.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295030EK3.06 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM

REFERENCE: PPM 5.0.10 rev 6 page 263

SOURCE: **NEW QUESTION** SRO T1, GP1, #19 RO T, GP2, #14

LO: 8383 – Given a list, identify the statement that describes the reason for entering PPM 5.1.1 RPV Control if wetwell level cannot be maintained below the SRVTPLL

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 gives the reason as stated in D. A is incorrect because there are no specific directions for low suppression pool level in 5.1.1. B is incorrect because during a LOCA, steam is exhausted through the DW floor downcomers, which exhaust below the SRV tailpipes. C is incorrect suppression pool volume below 19 feet 2 inches is not considered for ATWS.

COMMENTS:

The plant is in MODE 5 with the full core offloaded following an extended run at rated power. The normal cooling water supply to Fuel Pool Cooling Heat Exchangers has been lost.

Which of the following systems can be used as a backup cooling supply?

A. RCC Reactor Closed Cooling Water

B. CST Condensate Storage and Transfer

C. TSW Plant Service Water

D. SSW Standby Service Water

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295018AA1.01 3.3/3.4 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Backup Systems

REFERENCE: LO000202 rev. 10, page 16

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T1, GP2, #6 RO T1, GP2, #11

LO: 5371 – Given Fuel Pool Cooling and Cleanup System operating mode and various plant conditions, predict how key FPC/plant parameters will respond to the failure of the following support systems: a. Reactor Building Closed Cooling Water System

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: RCC is the primary source of cooling for FPC and not the backup. TSW and CST are not hard piped into the FPC heat exchangers. Standby Service Water is the system hard piped into the FPC heat exchangers for a backup system. D is correct.

COMMENTS:

The plant was operating at 98% power when a transient occurred. The following indications were received:

- Rod groups 1, 2, 3, and 4, power available lights (white) are out on P603 and P609
- Both RWCU Pumps have tripped
- AR-EX-1A trips
- NS4 Groups 2, 5, 6, and 7 (outboard) isolated

Which of the following is the reason for these indications?

A. Reference leg leak on MS-LT-26A (Wide Range MS-LR/PR-623A)

B. Variable leg leak on MS-LT-26D (Wide Range MS-LR/PR-623B)

C. Failed (low) undervoltage relay (27) on the EPA Breakers for RPS A.

D. Failed (low) undervoltage relay (27) on the EPA Breakers for RPS B.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 212000K6.05 3.5/3.8 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: RPS sensor inputs

REFERENCE: LO000161 rev. 11, pages 7 and 16, LO000126 rev 8, pages 8 and 10

SOURCE: **NEW QUESTION –** SRO T2, GP1, #5 RO T2, GP1, #8

LO: 5957 – Describe the function of these RPS EPA breaker components: d. Undervoltage light

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The indications given are for a loss of RPS making A and B incorrect. The loss of group 1, 2, 3, and 4 white lights and the outboard isolation indicate the loss of RPS A. C is correct.

COMMENTS:

An accident has occurred causing a Site Area Emergency due to offsite release rate. Rad Waste building ventilation has tripped off.

Which of the following is correct for these conditions?

Attempt to restart Rad Waste Building Ventilation to …

A. assure any release from the Radwaste Building is discharged through an elevated and monitored release point.

B. provide for Radwaste Building atmosphere recirculation and reduce the amount of radioactivity present.

C. cause a ground level release and limit the dispersion of radioactive material.

D. limit the intrusion of radioactive material from the Reactor Building.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295017AK3.02 3.3/3.5 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Plant Ventilation
REFERENCE:	PPM 5.0.10 rev. 6 page 302
SOURCE:	**BANK QUESTION LR00967– MODIFIED -** SRO T1, GP1, #15 RO T1, GP2, #9
LO:	8477 – Given a list, identify the statement that describes the purpose of restarting turbine building and Radwaste building HVAC during attempts to control offsite radioactivity release rates above the ALERT level.
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	B & C are incorrect because operation of RW Ventilation provides for an elevated exhaust, not recirculation of atmosphere. D is incorrect because RW building Ventilation is not designed to limit intrusion of radioactive material from the RB. A is correct as stated in the basis for PPM 5.4.1.
COMMENTS:	

The plant is in MODE 5 with fuel movement underway. The CRO notes both EDR-V-394 and 395, EDR-P-5 Discharge to Waste Collector Tank in Radwaste, have closed.

Which of the following caused these indications?

A. EDR-5 sump level High High

B. Drywell pressure 1.59 psig

C. ARM-RIS-23, CRD Pump Room, 215 mr/hr

D. Rx Building Exhaust Plenum 16 mr/hr

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295036EA1.04 3.1/3.4 10CFR55.41 - Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Radiation Monitoring

REFERENCE: LO000139 rev. 9, page 7

SOURCE: **BANK QUESTION – 2000 NRC EXAM**
SRO T1, GP, #15 RO T1, GP3, #4

LO: 5333 – List the isolation signal and setpoints for the following valves: EDR-V-394 & 395, ED-R-5 Sump Pump discharge isolation

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: EDR-V-394 and 395 isolate on an FAZ signal. D is correct.
COMMENTS:

The plant is operating at 99% power when an undervoltage occurs on SM-7 due to a spurious trip of BKR 1-7. CRA-FN-2A1 (MC-7B) has tripped from the loss of power.

Which of the following is correct concerning these conditions?

A. CRA-FN-2A1 must be manually restarted when MC-7B is repowered.

B. CRA-FN-2A1 auto starts 3 seconds following the undervoltage.

C. CRA-FN-2A1 auto starts 5.5 seconds after the undervoltage.

D. CRA-FN-2A1 auto starts 10 seconds after the undervoltage.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 223001K2.09 2.7/2.9 10CFR55.41 - Knowledge of electrical power supplies to the following: Drywell Cooling Fans

REFERENCE: LO000127 rev. 10, pages 30 & 31 LO000182 rev. 12 pages 41 and 88

SOURCE: **NEW QUESTION –** SRO T2, GP1, #13 RO T2, GP1, #19

LO: NO LO

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: MC-7B loses power during the undervoltage on SM-7. Since it is a critical power supply, it repowers and the fan starts 5.5 seconds later when B-7 closes. C is correct.

COMMENTS:

A plant startup is in progress. you have been given the direction to start the A Reactor Feed Pump.

What are the requirements for procedure usage for this evolution?

A. The procedure need not be present for this evolution if the operator assigned is familiar with the task.

B. The procedure must be present but steps can be skipped for more efficient operation of the plant if they are noted administratively on the front page of the procedure.

C. The procedure need not be present for this evolution but the steps of the procedure must be performed in order.

D. The procedure must be present and strict adherence to the procedure is required for operating the equipment.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.20 4.3/4.2 10CFR55.41, 43.5, & 45 Ability to execute procedure steps.

REFERENCE: SWP-PRO-01 rev. 3, page 11

SOURCE: **BANK QUESTION – 99 NRC EXAM –** SRO T3, #4 RO, T3, #2

LO: 6063 – State the requirement concerning whether or not approved plant procedures shall be used in the performance of plant activities.

RATING: L2

ATTACHMENT: N/A

COMMENTS:

JUSTIFICATION: The Company standard for procedure usage is STRICT ADHERENCE to the procedure. If the evolution is a complex operation and memory cannot be relied on then the procedure must be present. D is correct.

The plant is operating at 96% power with a normal electrical plant lineup when breaker 1-11 trips, breaker 21-11 auto closes and repowers SL-11 from SL-21.

Which of the following conditions caused these indications?

A. Undervoltage on SL-11

B. 86 lockout trip of breaker N1-1

C. Undervoltage on SM-1

D. Overcurrent on SL-11

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 262001A3.02 3.2/3.3 10CFR55.41 & 45 - Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Automatic bus transfer.

REFERENCE: LO000182 rev. 12, pages 62, 63, and 64

SOURCE: **BANK QUESTION –** SRO T2, GP1, #20 RO T2, GP2, #13

LO: 7767 – Describe the physical connection and/or cause and effect relationship between SL11 and : c. Breaker 21-11

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A, C, and D are incorrect because neither undervoltage on SL-11 or SM-1 nor overcurrent on SL-11 cause closure of 21-11. Only a lockout associated with TR-N1 or TR-S cause breaker 11-1 to trip and close breaker 21-11. B is correct.

COMMENTS:

The plant is in MODE 3 and the CRS has directed you to flush RHR to Radwaste in preparation for starting Shutdown Cooling per PPM 2.4.2.

Which of the following is correct for this condition?

RHR can be flushed to …

A. MWR-TK-23A/B Chemical Waste Tank

B. FDR-TK-9 Floor Drain Sample Tank

C. EDR-TK-4A/B Waste Sample Tanks

D. EDR-TK-5 Waste Surge Tank

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.3.10 2.9/3.3 10CFR55.43 – Ability to perform procedures to reduce excess levels of radiation and guard against personnel exposure.

REFERENCE: 82-RSY-0200-T6 page 2 and figure 1

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #12 RO T3, #9

LO: 5652 – Describe the normal flow path through the equipment drain processing system.

RATING: L2

ATTACHMENT: N/A

JUSTIFICATION: Of the listed tanks, only EDR-TK-5 is capable of receiving RHR flush water. D is correct.

COMMENTS: Removed section of answers/distracters that required knowledge of HP actions. Changed distracters to RW tanks and made the question a memory question.

The precautions for OSP-RSCS-C401 RSCS CFT Prior to Reactor Startup, prohibit selecting a bypassed control rod to verify RSCS Operability.

Which of the following describes the method used to verify that the selected control rod is available for RSCS Operability?

A. Depress the “RODS FULL IN/BYPASS” pushbutton to illuminate the BYPASS light. All bypassed control rods are indicated by a red LED.

B. Depress the “ALL RODS/FREE RODS” pushbutton to illuminate the FREE RODS light. All bypassed control rods are indicated by a flashing yellow LED.

C. If a bypassed control rod is selected on the rod select matrix, the RSCS INOP annunciator illuminates.

D. If a bypassed control rod is selected on the rod select matrix, the white position indication (XX-YY) does not illuminate on the full core display.

ANSWER: A

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	201004A4.01 3.4/3.5 10CFR41 & 45 - Ability to manually operate and/or monitor in the control room: System bypass switches
REFERENCE:	LO000160 rev. 9, page 7
SOURCE:	**BANK QUESTION – LO0898 –** SRO T2, GP2, #3 RO T2, GP2, #3
LO:	5807 – State the function of the following indications and controls on the RSCS operator console: RODS FULL IN/BYPASS.
RATING:	L2
ATTACHMENT:	NONE
JUSTIFICATION:	B is incorrect because a flashing yellow LED indicates the rod is selected on the rod select matrix. C is incorrect because selection of a bypassed control rod does not cause an INOP RSCS. D is incorrect because selection of a bypassed control rod does not cause the full core indication to change. A is correct because the purpose of the BYPASS position is to cause all bypassed control rods to indicate as stated.
COMMENTS:	This question has been reworded but has not changed enough for a modified question. It is the same as Bank Question LO00898.

An accident has occurred following an extended run at power. RCIC room temperature is approaching its maximum safe operating value.

Which of the following is correct concerning these conditions?

PPM 5.1.1 RPV Control must be entered to:

A. Ensure containment integrity.

B. Ensure adequate core cooling.

C. Reduce the dependence on RCIC for reactor level control.

D. Reduce the energy discharged to the Sec. Containment to decay heat level.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295032EK3.02 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Reactor SCRAM

REFERENCE: PPM 5.0.10 rev. 6, page 300

SOURCE: **NEW QUESTION –** SRO T1, GP2, #11 RO T1, GP3, #2

LO: 8457 – Given a list, identify the statement that describes the reason for entering PPM 5.1.1, RPV CONTROL if secondary containment parameters are approaching their Maximum Safe Operating Values and a primary system is discharging reactor coolant into secondary containment.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because entering PPM 5.1.1 does not ensure either containment integrity or adequate core cooling. C is incorrect again because entry into PPM 5.1.1 does not reduce the dependence on RCIC. D is correct as stated in PPM 5.0.10.

COMMENTS:

A plant startup is in progress at 234 psig and a RWCU blowdown in progress to the main condenser. Several alarms are received and upon investigation, the RO notes RWCU-FCV-33 closed. The RWCU lineup is otherwise normal.

Which of the following is the cause of these indications?

A. RWCU pressure upstream of RWCU-FCV-33 is 17 psig.

B. RWCU pressure downstream of RWCU-FCV-33 is 157 psig.

C. Reactor level is +5 inches.

D. Reactor level is –62 inches.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 204000A3.01 3.3/3.3 10CFR41 & 45 - Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Pressure downstream of pressure regulating valve

REFERENCE: LO000190 rev. 10, pages 8-11

SOURCE: **NEW QUESTION –** SRO T2, GP2, #6 RO T2, GP2, #6

LO: 5035 – List all RWCU System and filter Demineralizer isolations including setpoints and valves affected.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: System pressure must be either less than 5# upstream of 33 or greater than 140 downstream of 33 to isolate the valve. B would cause the valve to close and is correct. C and D are incorrect because reactor level of +5 inches has no effect on RWCU, while any level less than –50 inches causes a complete system isolation (V-1 & 4 close also).

COMMENTS:

The plant was operating at 99% power when a transient occurred. The following conditions exist:

P603 A7 drop 2.2 RPV PRESS HIGH TRIP	Illuminated
P603 A8 drop 2.2 RPV PRESS HIGH TRIP	Illuminated
P603 A8 drop 3.4 ½ SCRAM SYSTEM B	Illuminated
Reactor Pressure RFW-PI-605	1076 psig
Reactor Power	99%

Which of the following procedures is entered first?

A. ABN - PRESSURE

B. ABN - POWER

C. PPM 5.1.1 RPV Control

D. PPM 5.1.2 RPV Control - ATWS

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.45 3.3/3.6 10CFR55.43 – Ability to prioritize and interpret the significance of each annunciator or alarm

REFERENCE: PPM 1.3.1 rev 56, page 30

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #15 RO T3, #11

LO: 8044 – Given abnormal and annunciator response procedure steps that conflict with EOP steps being performed, determine which procedural steps take precedence.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: PPM 1.3.1 directs the priority/precedence of the Volume 5 procedures over the Vol. 4 Abnormals. With reactor pressure >1060 PPM 5.1.1 should be entered and the actions directed there taken. Even though this scenario is an ATWS, PPM 5.1.1 is the correct procedure. C is correct.

COMMENTS:
ex02012

A reactor startup is in progress with power at 60 on IRM range 2. Several alarms activate along with a Rod Out Block and a ½ Scram on RPS A.

Which of the following caused these indications?

A. Failure of DP-SO-A (24 VDC)

B. Failure of DP-S1-1A (125 VDC)

C. SRM A failed upscale.

D. IRM E ranged from range 2 to range 4.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215003K2.01 2.5/2.7 10CFR41 – Knowledge of electrical power supplies to the following: IRM channels/detectors

REFERENCE: LO000138 rev. 7, page 7 and LO000188 rev. 6, pages 27, 29, and 34

SOURCE: **BANK QUESTION – MODIFIED –** SRO T2, GP2, #10 RO T2, GP1, #10

LO: 7655 – Predict the effects a failure of 24VDC bus SOA will have on: c. IRM

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A failure of DP-SO-A causes a rod block and additionally a ½ scram from The IRM Inop. None of the other malfunctions cause a ½ scram. A is correct.

COMMENTS: This question was modified from LR01007, LO00532, and LX00058 and given a new number for this exam.

The plant was operating at 98% power when a Main Turbine trip causes a reactor scram. The lights in the control room go out for approximately 5.5 seconds and then some of the lights come back on.

Which of the following is correct for these conditions?

A. BKRs S-1, S-2, and S-3 have closed and are providing power.

B. BKRs N-1, N-2, and N-3 have closed and are providing power.

C. SM-7 and SM-8 are powered from DG-1 and DG-2.

D. SM-7 and SM-8 are powered from TR-B.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	295003AA2.05 3.9/4.2 10CFR55.41, 43 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Whether a partial or complete loss of AC power has occurred
REFERENCE:	LO000182 rev. 12, page 30
SOURCE:	**BANK QUESTION – 2001 NRC EXAM –** SRO T1, GP1, #2 RO T1, GP2, #4
LO:	5047 – State the open/closed status of the N and S breakers for SM1, 2, 3, SH5, and SH6 for: MT trip and TRS loss.
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	A is incorrect because the control room lights would not have gone out it the S BKRs had closed. B is incorrect be cause the N BKRs are the normal supply when the main turbine is on the line. C is incorrect because the light would have been out for at least 7 seconds if the DGs were powering the bus. D is correct because all control room lights, except those powered from MC-7C and 7E, and 8C and 8E would come back on when SM-7 and SM-8 were repowered.
COMMENTS:	

Which of the following systems is designed to maintain control room temperature at 75°F ± 3°F for personnel habitability if the **normal** cooling supply system **fails**?

A. Radwaste Chilled Water (WCH)

B. Control Room Emergency Chilled Water (CCH)

C. Standby Service Water (SW)

D. Turbine Service Water (TSW)

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 290003K3.01 3.5/3.8 10CFR41 & 45 - Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: Control room habitability.

REFERENCE: LO000201 rev. 9, pages 8, 19, and 20

SOURCE: **NEW QUESTION –** SRO T2, GP2, #13 RO T2, GP2, #18

LO: 5221 – State the purpose of each of the following system components: k. Control room emergency chillers

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because Radwaste Chilled water is the normal source for cooling. C is incorrect because Service Water will only maintain control room temperature at 104 °F for equipment. D is incorrect because TSW by itself does not cool the control room. B, Control room Emergency Chilled Water is the backup source of cooling water for personnel habitability. B is correct.

COMMENTS:

A plant startup is underway with the A2 sequence selected. All rod withdrawals in RSCS groups 1-4 have been completed.

Which of the following is the correct control rod density for this condition?

A. 25%

B. 50%

C. 75%

D. 100%

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 201004K5.02 3.1/3.3 10CFR55.41.5/45.3 - Knowledge of the operational implications of the following concepts as they apply to ROD SEQUENCE CONTROL SYSTEM: Sequences and groups

REFERENCE: LO000160 rev 10, page 7

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T2, GP2, #2 RO T2, GP2, #2

LO: 5806 – Explain the following term: Control rod density

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Completion of the 1st 4 RSCS rod groups places 50% of the rods in the full out position. This is 50% rod density. B is correct.

COMMENTS:

The plant is operating at 97% power when a radioactive spill occurs in the Radwaste Building. WEA-RIS-14 indicates 1.39E5 cpm and is stable. It will take at least 90 minutes before a plan is in place to begin work on the cleanup.

Which of the following is correct for these conditions?

A. Declare an Alert.

B. Declare a Site Area Emergency.

C. A four hour notification to the NRC is required.

D. A one hour notification to the NRC is required.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 288000 2.4.30 2/3.6 10CFR55.43.5 & 45: Plant Ventilation Systems – Knowledge of which events related to system operations/status should be reported to outside agencies.

REFERENCE: PPM 1.10.1 rev. 22, page 9, PPM 13.1.1 rev. 31, pages 19 & 35

SOURCE: **NEW QUESTION –** SRO T2, GP3, #3

LO: 6008 – State the notification requirements to State and Local Government agencies after declaring an Emergency Classification at Columbia.

RATING: H3

ATTACHMENT: **YES – PPM 13.1.1 rev. 31 pages 19 & 35 PPM 1.10.1 rev. 22, page 9**

JUSTIFICATION: The value given for WEA-RIS-14 is greater than the threshold for a UE if the release is expected to last more than an hour. Since it will take longer than 1 hour for a work plan, a UE must be declared. By declaring a UE, a 1 hour notification is required. D is correct.

COMMENTS:

The reactor had been operating at 99% power for the last six months. A plant transient occurred approximately two minutes ago causing a automatic reactor scram. The following conditions exist:

- Reactor level is 56 inches on the Narrow Range
- Reactor pressure is cycling between 1080 psig and 1094 psig
- HPCS and RCIC have not initiated
- No operator actions have been taken except the immediate scram actions

Which of the following is correct concerning these conditions?

The transient was caused by …

A. an APRM A INOP with a coincident loss of RPS-B.

B. a main generator trip from phase differential current.

C. a coincident loss of both SM-7 and SM-8.

D. a loss of SM-1.

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	295006AA2.04 4.1/4.1 10CFR55.41,. 43.5, and 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Reactor Pressure
REFERENCE:	LO000128 rev.8, page 4
SOURCE:	**NEW QUESTION –** SRO T1, GP1, #3
LO:	7682 – Describe the physical connection and/or cause and effect relationship between RPS and the following: MSIVs
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	From the information given it can be deduced that the MSIVs are isolated – pressure is at about 1090 psig and cycling. Reactor level is given at greater than +60 inches with neither HPCS nor RCIC having been initiated. A is incorrect because there would be no MSIV isolation at this point. B would cause a regular scram and no isolation. D would cause a loss of feedwater and an auto start of both HPCS and RCIC. A loss of both SM-7 and SM-8 would cause an immediate MSIV isolation and scram. C is correct.
COMMENTS:	After further research, the question stands as written. This is the response of the simulator under these conditions.

The plant has been operating at 99% power for the past 8 months. One SRV is leaking. Suppression pool temperature is 84°F and going up.

Why would the direction to place Suppression Pool Cooling in operation be given at this time?

To prevent…

A. exceeding the Tech Spec LCO of 90°F for Suppression Pool temperature.

B. exceeding the Tech Spec LCO of 110°F for Suppression Pool temperature.

C. a challenge to the wetwell to drywell interface integrity during the DBA LOCA..

D. a challenge to the SRV Tailpipe integrity during an emergency depressurization.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295013AK3.01 3.6/3.8 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool cooling operation

REFERENCE: PPM 5.0.10 rev. 6, page 248 TS 3.6.2.1 and bases pages B 3.6.2.1-1 & 2

SOURCE: **NEW QUESTION –** SRO T1, GP1, #10 RO T1, GP2, #7

LO: 6925 - Identify the basis for any LCO.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because 110° F is the SLC initiation value with the reactor at power. C and D are incorrect applications of other containment limits. A is the correct action to prevent exceeding the TS LCO.

COMMENTS:

The plant was operating at 75% power when a transient occurred causing reactor level to decrease to –7 inches.

Which of the following is correct concerning the initial direction from the CRS?

CRO-1 should be directed to…

A. P603, and Board A.
CRO-2 Board B and Board C
CRO-3 P601 and P602

B. P602, P603, and Board A.
CRO-2 Board B and Board C
CRO-3 P601

C. P602 and P603
CRO-2 Board A, Board B and Board C
CRO-3 P601

D. P603
CRO-2 Board A, Board B and Board C
CRO-3 P601, and P602

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.6 2.1/4.3 10CFR55.43.5/45.12/45.13 – Ability to supervise and assume a management role during plant transients and upset conditions.

REFERENCE: PPM 1.3.1 rev 56, page 28

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T3, #1

LO: 6092 – State the responsibilities of the Control Room Supervisor.

RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: PPM 1.3.1 states that the initial response for the CROs as those in B. B is correct.
COMMENTS:

The plant was operating at 99% power when an accident occurred. Core conditions have caused fuel damage. Radioactive conditions at the Site Boundary have caused an Alert to be declared.

Considering these conditions, which of the following procedures should be entered?

A. PPM 5.1.1 RPV Control

B. PPM 5.1.2 RPV Control ATWS

C. PPM 5.2.1 Secondary Containment Control

D. PPM 5.4.1 Radioactivity Release Control

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295017AA2.03 3.1/3.9 10CFR55.41, 43.5, & 45.8 - Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels

REFERENCE: PPM 13.1.1 rev. 31 pages 19 & 35 PPM 5.0.10 rev. 6 age 302

SOURCE: **NEW QUESTION –** SRO T1, GP1, #16 RO T1, GP2, #10

LO: 8017 – Given plant conditions, recognize an EOP entry condition and enter the appropriate flow chart

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The indication given for TEA-RIS-13 requires that an Alert be declared. PPM 5.4.1 entry is required when the release rate exceeds the Alert level. D is the correct answer.

COMMENTS: PPM 13.1.1 rev. 31 pages 19 & 35 were the handout pages for the original question. This reference is used for question 24 and needs to be included in the reference handout. This reference does not disclose to the examinee which procedure should be entered, only that an Alert should be declared which is given in the stem of the question. Therefore the reference has not been deleted from the reference handout.

A plant startup is underway. The following conditions exist:

APRM A, E, D	11%
APRM C and B	13%
APRM F	out of service – bypassed
IRM indications	25 to 35 on R10 for A, C, E, B, F, and H
IRM G	out of service – bypassed
IRM D	41 on R10
Reactor pressure	819 psig

Which of the following is the correct decision concerning these conditions?

A. Do not place the Mode Switch in RUN; a scram will occur from APRM C and D.

B. Do not place the Mode Switch in RUN; a MSL isolation will occur.

C. Place the Mode Switch in RUN; a mode change to RUN is allowed with at least 2 APRMs per trip system above 5%.

D. Place the Mode Switch in RUN; a mode change to RUN is allowed with at least 3 IRMs per trip system LE 40 on R10

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.2.2 4.0/3.5 10CFR55.45 – Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
REFERENCE:	LO000128 rev. 8, page21
SOURCE:	**BANK QUESTION – 2000 NRC EXAM–** SRO T3, #8 RO T3, #5 **WNP-2 LER 84-108**
LO:	NO LO
RATING:	H2
ATTACHMENT:	NO
JUSTIFICATION:	C and D are incorrect because the MS to run causes a full isolation and RX Scram. A is incorrect because the APRM scram setpoint is 15% with the MS in STARTUP. B is correct because MSL pressure of less than 831 psig in RUN causes a full MSIV isolation and reactor scram.

The plant is operating at 99% power with SM-4 de-energized and tagged out for maintenance. COND-P-2A trips due to a lockout.

Assuming no operator actions, which of the following is correct for these conditions?

A. RCIC initiates and injects until reactor level increases to +54 inches, RCIC-V-45 and 13 close until reactor level returns to -50 inches, where RCIC injects again.

B. HPCS initiates and injects until reactor level increases to +54 inches, HPCS-V-4 closes until reactor level returns to -50 inches, where HPCS injects again.

C. Recirc pumps runback to 30 hertz on low reactor water level and the feedpumps return reactor level to the normal operating band.

D. RHR injects and fills the RPV following an Automatic Depressurization System initiation.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295031EK2.04 4.0/4.1 10CFR55.41 & 45 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Reactor Core Isolation Cooling

REFERENCE: LO000180 rev. 11 pages 20 & 21

SOURCE: **NEW QUESTION –** SRO T1, GP1, #22 RO T1, GP1, #10

LO: 5714 – List the signal and setpoints that will cause a RCIC System initiation.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The loss of COND-P-2A causes a loss of feedwater (loss of feedpumps). Reactor level decreases to less than –50 inches and initiates RCIC. Because SM-4 is tagged out, HPCS does not start. RCIC returns level to +54 inches and RCIC-V-13 & 45 close. Reactor level does not decrease to the point of ADS/RHR initiation. A is correct.

COMMENTS:

You have assigned a maintenance person to repair a valve inside a valve room which is a posted RADIATION AREA. The job is projected to take four hours. The maintenance person has 1.985 rem TEDE this year.

Which of the following is correct for these conditions?

The maintenance person is allowed to work on this valve for a **maximum** of…

A. 1 hour

B. 2 hours

C. 3 hours

D. 4 hours

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.3.1 2.6/3.0 10CFR55.41.2/43.4/45.9/45.1 – Knowledge of 10 CFR 20 and related facility radiation control requirements

REFERENCE: GEN-RPP-07 rev. 3, page 7

SOURCE: **BANK QUESTION #98126 – MODIFIED –** SRO T3, #13

LO: 6013 – Define radiation area.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The Radiation area minimum threshold is 5 mr/hr. The **maximum** amount of time available to a worker for a 15 mr limit (2 rem admin limit) would be 3 hours in the area.

COMMENTS:

The plant is operating at 95% power with suppression pool level slowly and steadily going up. All systems are in the normal standby lineup.

Which of the following is the cause of the increasing suppression pool level?

A seat leak on …

A. RCIC-V-22 & 59 – Test Bypass

B. HPCS-V-23 – Full Flow Test

C. RHR-V-24B – Full Flow Test

D. LPCS-V-12 – Test Return

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 209002K1.03 3.0/3.0 10CFR55.41 & 45 - Knowledge of the physical connections and/or cause- effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: water leg pump

REFERENCE: LO000174 rev. 9, fig. 1 LO000192 rev. 9, fig. 1 LO000198 rev. 10, fig. 1 LO000180 rev. 11, fig. 13

SOURCE: **NEW QUESTION –** SRO T2, GP1, #3 RO T2, GP1, #6

LO: NO LO

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because both systems take the water leg pump suction from the suppression pool, and a leak would cause no net change in suppression pool level. A is incorrect because the RCIC is normally lined up with suction to the CSTs, but the Test Bypass line goes to the CSTs and not the suppression pool. B is correct because HPCS is normally lined up with the suction from the CSTs and a leak through the Full Flow Test valve would result in a net increase in suppression pool volume and level.

COMMENTS:

The plant is currently operating at 87% power. While performing a surveillance on Monday, MS-LIS-37A Level 1 input to ECCS Div 1 was determined to be inoperable at 1500. All required actions were taken. At 0300 on Tuesday, MS-LIS-37B, Level 1 input to ECCS Div 2 was found to be set at –144.5 inches.

Concerning these conditions, which of the following is correct?

A. No further action is required, MS-LIS-37B is set within tolerance.

B. Declare HPCS inoperable in 1 hour and place the channel in the trip condition within 24 hours.

C. Declare supported features inoperable within 1 hour, declare HPCS inoperable in 1 hour, and place the channel in the trip condition within 24 hours.

D. Declare supported features inoperable within 1 hour and place the channel in the trip condition within 24 hours.

ANSWER: D

QUESTION TYPE:	SRO
KA # & KA VALUE:	216000 2.2.25 2.5/3.7 10CFR55.43.2 – Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits: Concerning Nuclear Boiler Instrumentation.
REFERENCE:	TS 3.3.5.1 pages –1, -2, -8, -9, & -10 and bases pages B 3.3.5.1 –9, -10, -27, -28, & -29
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #9
LO:	10306 – With Tech Specs provided, determine LCO compliance for a given operational condition.
RATING:	H4
ATTACHMENT:	**YES -** TS 3.3.5.1 pages –1, -2, -8, -9, & -10 and bases pages B 3.3.5.1 –9, -10, -27, -28, & -29
JUSTIFICATION:	MS-LIS-37B is out of spec and it is the redundant initiation capability per TS. Because of this, the supported features must be declared inop within 1 hour and the channel must be placed in the trip condition within 24 hours. The HPCS initiation function is not affected so HPCS does not have to be declared inop. D is correct.
COMMENTS:	

The plant was operating at 99% power when loss of feedwater occurred. RCIC is in operation from its alternate source of water, and has restored reactor level to normal. The CRO reports RCIC discharge flow and discharge pressure are starting to oscillate and getting worse.

Which of the following describes the cause of these indications?

A. Low CST level

B. Low reactor water level

C. Low reactor pressure

D. Low suppression pool level.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	217000K5.01 2.6/2.6 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Indications of pump cavitation
REFERENCE:	General Physics BWR Gen. Fundamentals Chapter 2: Pumps pages 21 & 22
SOURCE:	**BANK QUESTION – 2000 NRC EXAM -** SRO T2, GP1, #11 RO T2, GP1, #16
LO:	7145 – Describe cavitation, including symptoms, effects on centrifugal pump operation and methods of prevention.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	GP Gen. Fun. Chapter 4 describes the effect of cavitation as an oscillation of discharge pressure. Cavitation is caused by a loss of NPSH. NPSH is the difference between the pressure on the suction side of the pump and the saturation pressure of the liquid being pumped. D is correct. It is the only possibility that lowers the suction pressure of the RCIC pump.
COMMENTS:	This question is similar to ex00053 which was used on the 2000 ILC exam. The question stem has been only changed enough to use it for the RCIC system. This is not considered a MODIFIED BANK question, but it was given a new number.

With the plant at 100% power, which of the following would indicate a leaking Main Steam Relief Valve?

A. Elevated SRV Acoustic Monitor indication.

B. Elevated SRV Tailpipe Temperature indication.

C. An SRV with a red indicating light on P628.

D. An SRV with a red indicating light on P631

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 239002K4.06 3.5/3.7 10CFR55.41 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Detection of valve leakage

REFERENCE: LO000129 rev. 8, pages 10 - 12

SOURCE: **NEW QUESTION –** SRO T2, GP1, #17 RO T2, GP1, #21

LO: 5528 – Describe the physical connection and/or cause and effect relationship between the SRVs and : Tailpipe thermocouples.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because these lights only indicate the solenoids are energized and not actual valve position or leakage. A is incorrect because the Acoustic Monitoring system was removed and is no longer in service. B is correct because an elevated temperature indicates steam flow through the valve.

COMMENTS:

The plant is in MODE 4. Maintenance was performed on the breaker for ROA-FN-1A and it is now ready for a test start. The following conditions exist:

The breaker is racked in and the fuses are installed.

The CRO places the control switch to start on both ROA-FN-1A and REA-FN-1A. He notes there is no control room indication of a start on ROA-FN-1A.

He places both control switches in the stop position and notes the green light is out for ROA-FN-1A and illuminated for REA-FN-1A.

The panel lights are checked and are found to be OK.

Concerning these conditions, which of the following is true?

A. Reactor building pressure increases to 4.7 inches of H2O and stabilizes.

B. Reactor building pressure increases to 4.0 inches of H2O and stabilizes.

C. Reactor building pressure increases to 4.7 inches of H2O and all Reactor Building Ventilation fans trip. Pressure returns to 0 inches of H2O

D. Reactor building pressure increases to 4.0 inches of H2O and all Reactor Building Ventilation fans trip. Standby Gas Treatment auto starts and maintains reactor building pressure at -.25 inches of H2O.

ANSWER: A
QUESTION TYPE: SRO/RO
KA # & KA VALUE: 290001A1.01 3.1/3.1 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: System Lineups
REFERENCE: 82-RSY-1000-T6 rev. 8, pages 5, 12, & 13 EWD-80E-001 WNP-2 LER 88-007
SOURCE: **NEW QUESTION –** SRO T2, GP1, #23 RO T2, GP2, #17
LO: 5677 – State the purpose of the alternate relief path damper.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: The information given indicates a blown or improperly installed trip fuse. The result of starting a fan with no trip circuit is a running fan that cannot be stopped from the control switch. In this case pressure would increase until both Div 2 fans trip and continue to increase pressure until the suction damper opened fully and limited pressure to 4.7 inches of H2O. A is correct.

The plant is operating at 95% power when several alarms are received on P603 including a Rod Out Block. On closer inspection the CRO notes that the UPSC, INOP, and DNSC indicating lights are illuminated on P603 for the A Rod Block Monitor.

Which of the following caused these indications?

A. Loss of RPS B.

B. Loss of RPS A.

C. An edge rod was selected.

D. RBM A was bypassed with the joystick.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215002K6.01 3.0/3.2 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM: RPS

REFERENCE: 82-RSY-0800-T1 rev. 8, pages 17 & 18

SOURCE: **NEW QUESTION –** SRO T2, GP2, #9 RO T2, GP2, #9

LO: 7667 – Predict the effects the following failures have on the RBM System: RPS

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The RBM System is divisionally powered from RPS. A loss of RPS-A causes the given indications. Selecting an edge rod or bypassing with the joystick does not result in a rod block. B is correct.

COMMENTS:

The plant is in Mode 5 with the refuel bridge (unloaded) over the spent fuel pool and refueling operations in progress. The Mode Switch is placed in the START/HOT STANDBY Position.

Considering these conditions, which of the following would cause a rod block annunciator in the control room?

A. Bypassing the SDV High Level Scram.

B. Withdrawal of a second control rod from the core.

C. The refuel bridge as it approaches the Reactor Cavity from the Spent Fuel Pool.

D. The main hoist loaded interlock as it picks up a fuel bundle in the Spent Fuel Pool.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.46 3.5/3.6 10CFR55.43 & 45 – Ability to verify alarms are consistent with the plant conditions

REFERENCE: LO000207 rev 8, page 25 & 26 LO000148 rev. 10 pages 12 & 13

SOURCE: **BANK QUESTION - MODIFIED – 2001 NRC EXAM –** SRO T3, # 16 RO T3, #12

LO: 5360 – List and explain all the conditions pertaining to fuel handling equipment that can cause a control rod block when the reactor mode switch in either the REFUEL position or the START/HOT STBY position.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Placing the Mode Switch to START/HOT STANDBY causes a rod block when the refuel bridge is near the core. C is correct. A, B, and D are incorrect because these actions do not cause a rod block with the Mode Switch in STARTUP.

COMMENTS:

Preparations are in progress for a plant startup. The following conditions exist:

- RCIC is out of service for repair for the next two hours
- MS-LIS-24C (reactor level 3) is set at 9.1 inches
- All other plant equipment and conditions are normal

You have been directed to begin the startup.

Which of the following is correct concerning these conditions?

A. The Mode Switch cannot be placed in STARTUP until the MS-LIS-24C setpoint is returned to an allowable value.

B. The Mode Switch cannot be placed in STARTUP until RCIC is returned to service.

C. Place the Mode Switch in STARTUP, MS-LIS-24C is not required to be within specifications in MODE 2.

D. Place the Mode Switch in STARTUP, RCIC is not required to be in service until steam dome pressure is GT 150 psig.

ANSWER: A

QUESTION TYPE:	SRO
KA # & KA VALUE:	295009 2.2.24 2.6/3.8 10cfr55.43.2 & 45 - Ability to analyze the affect of maintenance activities on LCO status.
REFERENCE:	PPM 4.603.A7 rev. 28, page 15 Tech Spec 3.3.1.1 pages –1 & -8 TS 3.0.4 pages 3.0-1 & -2
SOURCE:	**NEW QUESTION –** SRO T1, GP1, #6
LO:	10308 – With Tech Specs provided and given an operational condition, determine if a Mode change or other change in plant condition is allowed.
RATING:	H3
ATTACHMENT:	**YES -** PPM 4.603.A7 rev. 28, page 15 Tech Spec 3.3.1.1 pages –1 & -8 TS 3.0.4 pages 3.0-1 & -2
JUSTIFICATION:	A is correct because placing the Mode Switch in STARTUP at this point relies on an action statement for the out of spec level switch (MS-LIS-24C), which is not allowed by TS 3.0.4. B is incorrect because RCIC is not required to be operable until steam dome pressure is GT 150 psig. C and D are incorrect because the Mode Switch cannot be placed in STARTUP until MS-LIS-24C calibration is within spec.

Which of the following is designed to prevent secondary containment over-pressurization during a postulated piping break in the area between the Drywell and the Turbine Building?

A. Standby Gas Treatment.

B. Reactor Building Ventilation.

C. Reactor Building Blowout Panels.

D. Main Steam Tunnel Blowout Panels.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295035EK1.02 3.7/4.2 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Radiation release

REFERENCE: LO000139 rev. 9, page 6

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** SRO T1, GP2. #14 RO T1, GP3, #3

LO: 7003 – State the actions that occur on a Main Steam tunnel High Pressure.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: An over pressurization in the Main Steam Tunnel results pressure relief through the MST Blowout panels. This relief minimizes the damage to the Secondary Containment and limits radioactive releases from any other part of the Sec. Cont. D is correct.

COMMENTS:

COLUMBIA GENERATING STATION WRITTEN EXAMINATION OCT. 2002

QUESTION # 41 EXAM KEY 10/04/2002

ex02026

A reactor scram just occurred. Several control rods do not indicate full-in

How are these control rods identified on the Rod Worth Minimizer?

A. A list containing each rod not full in by it's numerical position (i.e. 30-31).

B. Both a list containing each rod not-full-in by it's numerical position (i.e. 30-31) and a full core display with rods not-full-in indicated by • • (2 dots).

C. A full core display map with rods full-in indicated by ++ and rods not full in indicated by >>.

D. A full core display map with rods full-in indicated by >> and rods not full in indicated by ++.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295015AK2.05 2.6/2.8 10CFR55.41 & 45 - Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Rod Worth Minimizer

REFERENCE: LO000154 rev. 10, pages 12, 13, 16, & fig. 10

SOURCE: **NEW QUESTION –** SRO T1, GP1, #12 RO T1, GP1, #6

LO: 5908 – Given a set of RWM displays, explain the following symbols:

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is correct because any control rod not full in following a scram automatically is indicated on the RWM by a numerical listing of it's position in the core. B, C, and D are all incorrect because ·· indicated a rod inserted past position 02 in the scram mode, ++ indicates a rod fully withdrawn, and > > is not a RWM indication.

COMMENTS:

The plant is shutdown. RHR has been lined up for suppression pool cooling and RCIC is running for level control, when S2-1 is de-energized.

Which of the following describes the effect on plant operation?

A. CN-V-65 auto closes and isolates the N2 supply to the containment.

B. TO-P-BOP main turbine bearing oil pump trips and trips the MT turning gear.

C. RCC-V-6 auto closes and isolates the non-drywell RCC loads.

D. RCIC will continue to operate and provide level control.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295004AK2.03 3.3/3.3 10CFR55.41 & 45 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: DC bus loads

REFERENCE: ABN-ELEC-250VDC rev. 0, page 5

SOURCE: **BANK QUESTION – 98 NRC EXAM** SRO T1, GP2, #3 RO T1, GP2, #5

LO: 7657 – Predict the effect a failure of 250 VDC bus S2-1 will have on RCIC.

RATING: H3 H4

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because CN-V-65 is blocked open. B and C are both incorrect because neither pierce of equipment is powdered from 250 VDC. D is correct because the loss of S2-1 will not prevent RCIC from operating.

COMMENTS:

The plant is operating at 78% power when a total loss of DP-S1-2 occurs. Shortly thereafter a DBA LOCA occurs.

Which of the following is correct concerning these conditions?

A. HPCS-P-1 does not start due to a loss of power to the initiation logic and breaker.

B. RHR-P-2B does not start due to a loss of power to the initiation logic and breaker.

C. RHR-P-2A starts but does not inject due to the loss of power to the injection valve.

D. LPCS-P-1 starts but does not inject due to the loss of power to the injection valve.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 203000K2.03 2.7/2.9 10CFR55.41 - Knowledge of electrical power supplies to the following: Initiation logic

REFERENCE: EWD-9E-002, 47E-003, & 47E-003A

SOURCE: **NEW QUESTION –** SRO T2, GP1, #2 RO T2, GP1, #3

LO: 7653 – Predict the effects of a failure of 125VDC bus S1-21 will have on RHR.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because HPCS logic is from Div 3 power. C and D are both incorrect because the injection valve is AC powered and will open. B is correct because of the loss of control power.

COMMENTS:

The reactor is at rated power with TIP Channel C inserted for LPRM calibration when a loss of SM-1 occurs.

Assuming no operator action, which of the following is correct?

A. The TIP drive continues to insert the detector to the Core Top Limit and completes the Tip trace. The detector then withdraws into the shield chamber and the ball valve closes.

B. The squib valve for TIP channel C fires immediately isolating the drive mechanism.

C. The inserted TIP detector withdraws into the shield chamber, the ball valve closes, and TIP-V-15, Tip Purge Isolation Valve closes.

D. The inserted TIP detector stops until power is restored to SM-1 and then completes the TIP trace.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215001K4.01 2.9/3.3 10CFR55.41 - Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation.

REFERENCE: LO000155 page 13 and 16

SOURCE: **BANK QUESTION – 2000 NRC EXAM** - SRO T2, G3, #2 RO T2, G3, #1

LO: 6989 – Explain the TIP system response to a LOCA signal.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the drive does not continue for a complete trace. B is incorrect because the squib does not fire on and isolation signal. D is incorrect because the loss of power has no direct affect on the TIP drive.

COMMENTS:

The plant is in MODE 2 with a control rod withdrawal in progress. An LPRM failure has caused an I&C technician to take the LPRM mode selector switch out of OPERATE. This results in 13 LPRMS assigned to APRM A.

Which of the following is correct for these conditions?

A. A rod block is applied by the RMCS. Rod withdrawal can continue following bypass of APRM A.

B. A rod block is applied by the RMCS. Rod withdrawal can continue following bypass of the failed LPRM.

C. No rod block is applied by the RMCS for this failure.

D. No rod block is applied by the RMCS until rod movement is attempted.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215005K3.03 3.3/3.3 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the APRM/LPRM will have on following: Reactor Manual Control System

REFERENCE: 82-RSY-0500-T2 rev. 8, page18 LO000148 rev. 10, page 14

SOURCE: **NEW QUESTION –** SRO T2, GP1, #8 RO T2, GP1, #13

LO: 5795 – Identify the conditions that will cause rod blocks: d. RMCS Block

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Less than 14 LPRMs in OPERATE to an APRM results in an APRM Inop and a control rod out block applied by the RMCS. Taking the MODE switch out of operate for the failed LPRM caused the APRM inop. A is correct.

COMMENTS:

The plant was operating at 75% power when an ATWS with a full MSIV isolation occurred. Suppression chamber pressure is 27 psig and up, with drywell pressure trending up approximately 0.5 psig less than suppression chamber.

Which of the following is correct concerning these conditions?

A. The wetwell to drywell vacuum breakers are covered with water.

B. The wetwell to drywell vacuum breakers are working correctly.

C. A drywell downcomer has failed below the water level in the suppression pool.

D. The drywell to suppression chamber interface has ruptured.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 223001A3.02 3.4/3.4 10CFR55.41 & 45 - Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: Vacuum breaker relief valve operation.

REFERENCE: LO000127 rev. 10, page 8

SOURCE: **NEW QUESTION –** SRO T2, GP1, #14 RO T2, GP1, #20

LO: 5636 – State the purpose of the following major components: wetwell to drywell vacuum breakers.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The information given indicates a leak in an SRV downcomer. Suppression chamber pressure is increasing above drywell pressure by 0.5 psig which indicates that the wetwell to drywell vacuum breakers are working correctly. B is correct.

COMMENTS:

Reactor building ventilation has failed. The CRO has started Standby Gas Treatment by placing the control switch for SGT-EHC-1B-2 in the ON position.

Which of the following is correct concerning these conditions?

A. Room discharge valve (SGT-V-4A2), auto opens.

B. The associated Elevated release discharge valve (SGT-V-5A1), auto closes.

C. The incoming gas temperature is raised by the electric heaters to reduce humidity and increase efficiency of the moisture separators.

D. The incoming gas temperature is raised by the electric heaters to reduce humidity and increase efficiency of the charcoal adsorbers.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 261000A1.07 2.8/2.9 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: SGTS train temperature;.

REFERENCE: 82-RSY-0400-T3 rev. 10, pages 3-5 & 10

SOURCE: **NEW QUESTION –** SRO T2, GP1, 19 RO T2, GP1, #25

LO: 5825 – State the purpose of the SGT electrical heating coils.
5829 – Describe the SGT System response to operation of the heater control switch.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are incorrect because these valves act in opposite direction as given. D is incorrect because the moisture separators are upstream of the heaters and take moisture out prior to the heating coils. D is the stated purpose of the electric heating coils.

COMMENTS:

What is the purpose of sequencing loads on DG-1 and 2?

A. Prevents the possibility of over speeding the diesel.

B. Prevents a generator loss of excitation trip.

C. Reduces the automatically connected loads.

D. Reduces the tendency for the diesel engine to run under “souping” conditions.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 264000K5.06 3.4/3.4 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Load sequencing

REFERENCE: LO000200 rev. 8, pages 40 & 41

SOURCE: **NEW QUESTION –** SRO T2, GP1, #21 RO T2, GP1, #27

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because load sequencing would have no effect on the possibility of over speeding the DG or preventing a loss of excitation trip. D is incorrect because souping is prevented by not operating the DG for long periods of time under lightly loaded conditions. C is correct as given in the systems text.

COMMENTS:

Which of the following design features acts first to limit Secondary Containment overpressure?

A. Opening of ROA-AD-5, ROA suction relief damper.

B. Trip of the operating Standby Gas Treatment fans.

C. Secondary Containment (606 elevation) blowout panels.

D. Trip of reactor building supply and exhaust fans.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 290001K4.02 3.4/3.5 10CFR55.41 - Knowledge of SECONDARY CONTAINMENT design feature(s) and/or interlocks which provide for the following: Protection against over pressurization.

REFERENCE: 82-RSY-1000-T6 rev. 8, pages 5 & 12 – 13 LO000139 rev.9, page 4

SOURCE: **NEW QUESTION –** SRO T2, GP1, #22 RO T2, GP2, #16

LO: 5678 – Explain how Reactor Building pressure is maintained by the Reactor Building HVAC system including the controller setpoint.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because there is no SGT trip on high pressure. C and D are both incorrect because the blowout panels and the fan trips occur at a higher pressure than the suction damper actuations. A is correct.

COMMENTS:

The plant is operating at 99% power when both ROD ACCUMULATOR TROUBLE and SCRAM VALVE PILOT AIR HEADER PRESSURE LOW annuciators alarm simultaneously.

Which of the following is correct for this condition?

Annunciator…

A. ROD ACCUMULATOR TROUBLE should be responded to first because the setpoint for this alarm is an LCO entry.

B. ROD ACCUMULATOR TROUBLE should be responded to first because this alarm indicates an unscramable control rod.

C. SCRAM VALVE PILOT AIR HEADER PRESSURE LOW should be responded to first because a full reactor scram may occur without operator response.

D. SCRAM VALVE PILOT AIR HEADER PRESSURE LOW should be responded to first because control rods may not scram if scram air header pressure decreases too far.

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	201001 2.4.45 3.3/3.6 10CFR55.43.5 & 45 – Concerning the CRDH system, the ability to prioritize and interpret the significance of each annunciator or alarm.
REFERENCE:	PPM 4.603.A7, drop6-7, rev. 28 page 70 PPM 4.603.A8, drop 6-4, rev. 18, page 58
SOURCE:	**NEW QUESITON –** SRO T2, GP2, #1
LO:	NO LO
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	A and B are both incorrect because this annunciator is neither an LCO entry nor is it an indication that the control rod will not scram. D is incorrect because without scram pilot valve air header pressure, the control rods will scram. C is correct: with decreasing scram pilot air header pressure, a scram will occur with no operator action and no change in the leakage.

The plant is operating at 99% power when a fault occurs. The RWM will no longer display the current control rod position information.

According to PPM 2.1.4 Rod Worth Minimizer, which of the following should cause the RWM to automatically restart?

Repower of…

A. RPS-A

B. US-PP

C. IN-2

D. IN-3

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 201006A2.01 2.5/2.8 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those: Power supply loss

REFERENCE: LO000154 rev. 10, page 25 PPM 2.1.4 Rod Worth Minimizer rev. 12, page 12

SOURCE: **NEW QUESTION –** SRO T2, GP2, #4 RO T2, GP2, #4

LO: 5916 – Describe the physical connection and/or cause and effect relationship between RWM and: e. IN1

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A loss of power causes the indications given. US-PP is the power supply to the RWM. Restoration of US-PP caused the auto start of the RWM. B is the correct answer, none of the distracters have an effect on the RWM.

COMMENTS:

The plant is at 25% power following a maintenance outage for work in the drywell. The Primary Containment is being inerted when the EO reports that the Liquid Nitrogen Storage Tank Level is at 49 inches and down slow on CN-LIS-1. ADS header pressure has been 149 psig for the last four minutes.

Which of the following is correct for these conditions?

A. The CIA programmers placed their respective banks in service but stopped at step 1 and CIA-V-39A and 39B remained open.

B. The CIA programmers placed their respective banks in service and CIA-V-39A and 39B remained open.

C. The CIA programmers placed their respective banks in service but stopped at step 1 and CIA-V-39A and 39B have isolated.

D. The CIA programmers placed their respective banks in service and CIA-V-39A and 39B have isolated.

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.3.9 2.5/3.4 10CFR55.43 – Knowledge of the process for performing a containment purge.
REFERENCE:	PPM 2.3.1 rev 39, page 31 LO000156 rev. 6, page 8
SOURCE:	**BANK QUESTION 2000 NRC EXAM – WNP-2 LER 90-022 –** SRO T3, #13 RO T3, #7
LO:	5150 – State the 3 signals that cause auto initiation of the nitrogen bottle bank programmer to maintain system pressure.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	LER 90-022 documents the problem of CIA header isolation and inoperability due to inerting the containment with low levels of nitrogen in the nitrogen storage tank. CIA header pressure drops and causes CIA-V-39A and 39B to isolate. D is correct.
COMMENTS:	Question rewritten to make it closed reference. Removed reference from handout. Changed question to ADS pressure.

The plant is operating at 33% power when the CRO arms and depresses both MSIV ISOLATION LOGIC B and D pushbuttons

Which of the following is correct for these conditions?

A. All inboard and outboard isolation valves close; MSIVS stay open.

B. All inboard and outboard isolation valves close; MSIVS close.

C. The inboard and outboard MSIVs close.

D. There are **no** isolation valve closures.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.4.21 3.7/4.3 10CFR55.43 & 45 – Knowledge of the parameters and logic used to assess the status of safety functions: Radioactivity Release control

REFERENCE: LO000173 rev. 9, page 9

SOURCE: **BANK QUESTION LX00747 – MODIFIED –** SRO T3, #17 RO T3, #13

LO: 5600 – Explain the response to arming and depressing the MSIV isolation logic pushbuttons on P601.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Depressing the B pushbutton, causes an Outboard isolation. Depressing D causes an Inboard isolation. The combination of the 2 pushbuttons does not cause anything other than the combined results of press each individually. A is correct. There is no MSIV motion.

COMMENTS:

The reactor was operating at 78% power coming out of a refueling outage. A large steam leak in the drywell caused the following plant conditions:

Wetwell level	39 feet
Drywell pressure	30 psig
Reactor pressure	214 psig
Reactor level	-145 inches and stable

RCIC tripped several minutes ago.

Which of the following caused the RCIC trip?

A. Low reactor level.

B. Isolation from low reactor pressure.

C. Low suction pressure.

D. High exhaust pressure.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.7 3.7/4.4 10CFR55.55.43 – Ability to evaluate plant performance and make operational judgments base on operating characteristics/ reactor behavior / and instrument interpretation.

REFERENCE: PPM 5.0.10 rev 6, page 70

SOURCE: **BANK QUESTION – #EX00099 – 2000 NRC EXAM–** SRO T3, #3 RO, T3, #1

LO: 5722 - List the RCIC isolation signals and setpoints.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: A is incorrect because there is no trip on low reactor level. Low pressure isolation has not yet been reached for B. C is incorrect because suction pressure would be relatively high from the conditions given. D is correct based on the explanation of Caution 4 in the EOPs.
COMMENTS:

PPM 5.1.2 has been entered due to ATWS conditions. Level/power conditions exist.

Which of the following is true?

Reactor water level is lowered to .....

A. lower the height of the fluid columns, thereby reducing natural circulation and reactor power

B. reduce inlet sub-cooling, which reduces the void fraction and reactor power

C. reduce the rate of steam removal from the core, which reduces the void fraction and reactor power

D. increase natural circulation driving head and flow through the core to reduce reactor power

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295031EK1.03 3.7/4.1 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power.

REFERENCE: PPM 5.0.10 Rev 6, page 155

SOURCE: **BANK QUESTION – ex98001 – 98 NRC EXAM –** SRO T1, GP1, #21 RO T1, GP1, #9

LO: 7498 – Explain the causes of natural circulation in BWRs.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B and C have correct initial conditions, inlet sub cooling and steam production both decrease, but the void fraction would INCREASE from these effects, not decrease. In D, natural circulation head would decrease, not increase.

COMMENTS:

A fire has occurred in the area of Reactor Narrow Range level instrumentation tubing in the reactor building.

Which of the following is correct concerning these conditions?

Reactor water level instrumentation operability must be determined because…

A. heating in the variable leg results in unstable and higher indicated level than actual.

B. heating in the reference leg results in unstable and higher indicated level than actual.

C. water spray from the fire suppression systems causes failures of reactor level instrumentation from operation in non-qualified atmosphere.

D. smoke from the fire causes failures of reactor level instrumentation from operation in non-qualified atmosphere.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 600000AK3.04 2.8/3.4 - Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: actions contained in the abnormal procedures for plant fire on sight.

REFERENCE: ABN-FIRE rev. 3 page 12

SOURCE: **BANK QUESTION – LX00017 –** SRO T1, GP2, #17 RO T1, GP2, #19
LO: 7166 – Given a potential failure mode for a differential pressure cell used for level indication, describe how indicated level will be affected.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: Any heat source place on the reference leg of a level instrument can cause false hi level indications due to the reduction in density of the ref. leg. Heating in the variable leg would cause a false lower indicated level. Smoke and water in the atmosphere would have no impact on the level instrument. B is correct.
COMMENTS:

A plant startup is in progress. The following conditions exist:

- All IRMs are inserted and indicate downscale on Range 1
- All SRMs are inserted - the RETRACT PERMIT light is NOT illuminated for SRM A.
- The mode switch is in the STARTUP/HOT STANDBY position

Which of the following is correct?

If SRM A is selected and a withdraw signal is applied, the detector…

A. does not move and a rod block is generated.

B. does not move and RETRACT NOT PERMITED annunciator illuminates.

C. retracts and a ½ scram is generated on RPS A.

D. retracts and a rod block is generated.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 215004A4.06 3.2-3.1 10CFE55.41 & 48 - Ability to manually operate and/or monitor in the control room: SRM back panel switches, meters, and indicating lights.

REFERENCE: LO000132 rev. 9, page 18 and 19

SOURCE: **NEW QUESTION –** SRO T2, GP1, #6 RO T2, GP1, #12

LO: 5942 – Explain what is meant by SRM RETRACT PERMIT light when energized.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because the detector starts to withdraw as soon as the withdraw signal is initiated. C is incorrect because there is no ½ scram initiated. D is correct. At less than 100 counts, a rod block is generated if a detector is withdrawn.

COMMENTS:

The plant was operating at 99% power when a LOCA occurred. The CRO reports the following conditions:

- RPV pressure = 30 psig
- Drywell temperature = 300°F
- All level indicators are swinging 20 to 40 inches
- Control rod 30-31 is stuck full out

As the CRS, which of the following procedures should you enter?

A. 5.1.3 Emergency RPV Depressurization

B. 5.1.4 RPV Flooding

C. 5.1.5 Emergency RPV Depressurization - ATWS

D. 5.1.6 RPV Flooding - ATWS

ANSWER: B

QUESTION TYPE: SRO
KA # & KA VALUE: 216000 2.1.6 10CFR43.5 & 45 – Ability to supervise and assume a management role during plant transients and upset conditions (Nuclear Boiler Instrumentation)
REFERENCE: PPM 5.1.1 RPV Control
SOURCE: **NEW QUESTION –** SRO T2, GP1, #7
LO: 8017 – Given plant conditions, recognize an EOP entry conditions and enter the appropriate flow chart.
RATING: H3
ATTACHMENT: **YES** - PPM 5.1.1 RPV Control
JUSTIFICATION: C and D are both incorrect because one control rod full out does not meet the criteria for an ATWS. With erratic level indications and conditions outside the bounds of saturation conditions for level indications, the direction is given to exit both the level and pressure legs of PPM 5.1.1 and enter PPM 5.1.4. B is correct.
COMMENTS:

The plant is operating at 97% power with a discharge from the Waste Collector Tank to the Circ. Water Blowdown line underway. Process Rad Monitor FDR-RIS-606 (Radwaste Effluent) fails downscale.

Which of the following is correct concerning these conditions?

A. The discharge may continue for up to 30 days provided grab samples are collected and analyzed for radioactivity of at least 10^{-7} microcurie/ml, at least once every 12 hours.

B. The discharge may continue for up to 30 days provided that the discharge flowrate is estimated at least once every 4 hours during the release.

C. Stop the discharge. The discharge may continue when 2 independent samples have been analyzed and 2 technically qualified members of the plant staff have independently verified the release calculations and the discharge valve lineup.

D. Stop the discharge. The discharge may continue when a temporary monitor has been installed and the monitor calibration has been verified by analysis of 2 independent batch samples.

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.3.3 1.8/2.9 10CFR55.43.4 – Knowledge of SRO responsibilities for auxiliary system that are outside the control room.
REFERENCE:	ODCM 6.1.1 table 6.1.1.1-1
SOURCE:	**BANK QUESTION – 2000 NRC EXAM –** SRO T3, #9
LO:	NO LO
RATING:	H3
ATTACHMENT:	YES - ODCM 6.1.1 table 6.1.1.1-1, PPM 4.602.A5.6-6
JUSTIFICATION:	A and B are incorrect because they both allow the discharge to continue and the actions given are for the SW monitors and for the flowrate monitor of Rad Waste. D is incorrect because there is no action allowing the use of a temporary monitor in the place of FDR-RIS-606. C is correct. This is the action given in the ODCM.
COMMENTS:	I need some more direction on this one from Ryan. The direction to stop the discharge is on one page and the actions are determined after referring to another table then referring to specific actions. Talk to Ryan. Made A and B action 101 and 102 of table 6.1.1.1-1.

The plant is in MODE 2 following a refuel outage. The following conditions exist:

Reactor power = 2%
RCIC is in Full Flow Test operation
Suppression pool temperature = 106°F

Which of the following is correct concerning these conditions?

A. Stop RCIC immediately to preserve the heat absorption capability of the suppression pool.

B. Place the Mode Switch in SHUTDOWN immediately to preserve the heat absorption capability of the suppression pool.

C. Reduce suppression pool temperature to less than 90°F within 1 hour to prevent exceeding the HCTL during a LOCA.

D. Reduce reactor power to less than 1% within 1 hour to prevent exceeding the HCTL during a LOCA.

ANSWER: A

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.4.49 4.0/4.0 10CFR55.41, 43.2, & 45 – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
REFERENCE:	Tech Spec 3.6.2.1 and TS Bases B 3.6.2.1
SOURCE:	**NEW QUESTION** SRO T3, #13
LO:	6926 – Given a set of plant conditions, be able to identify and state from memory the Technical Specification actions required to be taken within 15 minutes or less.
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	B is incorrect because at this temperature, there is no requirement to place the Mode Switch in SHUTDOWN. C is incorrect because there is no requirement to reduce suppression pool temperature to less than 90° within 1 hour for any reason. D is incorrect because at this temperature there is no requirement to reduce power to less than 1%. A is correct as state in the action statement and the Bases.
COMMENTS:	

ex00011

Which of the following describes the Tech Spec basis for the End of Cycle Recirc Pump Trip (EOC RPT)?

A. Control rod insertion may not initially add enough negative reactivity to overcome the positive reactivity added by the pressure increase from a turbine trip.

B. Control rod insertion initially adds positive reactivity late in core life that must be compensated for by the trip of both Recirc pumps.

C. Recirc Pumps must be tripped to reduce the positive reactivity addition from the turbine trip and prevent exceeding MAPRAT.

D. Recirc Pumps must be tripped late in core life to minimize the effect of all control rods withdrawn to the full out position and prevent exceeding the LHGR

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295007AA2.02 4.1/4.1 10CFR55.41, 43.2 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor Power

REFERENCE: TS Bases B3.3.4.1 EOC-RPT Instrumentation

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T1, GP1, #5

LO: 6925 – Identify the basis for any LCO.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: TS Bases states the EOC-RPT trip is designed to overcome the lack of negative reactivity in the first few feet of control rod insertion on a scram. This lack of negative reactivity is cause by EOC flux shape. The Recirc pumps are tripped to supplement the control rod negative reactivity on a turbine trip. A is correct.

COMMENTS:

ex00062

The reactor is operating at rated conditions.

Which of the following describes how CRDM graphitar seal embrittlement is prevented?

A. The CRD Mechanism is monitored for temperature by a thermocouple in the instrument tube and maintained less than 250°F.

B. The CRD Mechanism is monitored for temperature by a thermocouple in the outer tube and maintained less than 250°F.

C. Cooling water from CRDH is supplied to the P-over port at a high enough flow rate to ensure sufficient cooling of the CRD Mechanism.

D. Cooling water from CRDH is supplied to the outside of the thermal sleeve at a high enough flow rate to ensure sufficient cooling of the CRD Mechanism.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 214000K4.02 2.5/2.5 10CFR55.41 - Knowledge of ROD POSITION INFORMATION SYSTEM design feature(s) and/or interlocks which provide for the following: Thermocouple

REFERENCE: LO000137 pages 10 and 13

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T2, GP2, #8
RO T2, GP2, #8

LO: 5217 – Explain how rod temperatures are monitored.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: C and D are both incorrect because they describe an incorrect flow path. B is incorrect because the thermocouple is located inside the instrument tube not the base of the stop piston.

COMMENTS: Changed the reference to the base of the stop piston to the outer tube to make B more clearly incorrect.

The plant is in MODE 3 with a reactor level reduction occurring through the A RHR Heat Exchanger Vents to the Suppression Pool. The Suppression Pool high level alarm has been sealed in for a period of time when the HPCS Suction Switchover occurs.

Which of the following describes the required action for these conditions?

A. Be in MODE 4 in 36 hours.

B. Restore suppression pool water level to within limits and restore HPCS suction to the CSTs within 2 hours.

C. Enter PPM 5.2.1 Primary Containment Control and restore suppression pool water level to within limits in 12 hours.

D. Enter PPM 5.2.1 Primary Containment Control and restore suppression pool water level to within limits in 2 hours.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 2.1.33 3.4/4.0 10CFR55.43 – Ability to recognize indications for system operating parameters which are entry level conditions of technical specifications.

REFERENCE: TS 3.6.2.2 PPM 5.0.10 rev. 6, page 246

SOURCE: **BANK QUESTION – 2000 NRC EXAM –** SRO T3, #5 RO T3, #3
WNP-2 LER 91-015

LO: 5429 – List the automatic initiations and interlocks associated with HPCS system components.

RATING: H3

ATTACHMENT: YES – TS 3.6.2.2

JUSTIFICATION: The high level suppression pool suction switchover for HPCS occurs at +3 inches in the suppression pool. This level requires entry into PPM 5.2.1 Primary Containment Control. This is also above the LCO for suppression pool water level. TS require that suppression pool level be restored to within limits in 2 hours. There is no requirement to return the HPCS suction valves to CST suction. D is correct.

COMMENTS: After review, Ryan said make distracter A “Be in MODE 4 in 36 hours”. This is clearly incorrect.

A plant startup is in progress following a refueling outage. Primary containment is being inerted with nitrogen supplied through the high flow inerting supply header. Drywell pressure has increased to 1.78 psig.

Which of the following is correct for these conditions?

A. Inerting continues as established.

B. CSP-V-106 and 107 (N2 tank isolations) close to isolate the high flow header.

C. CSP-V-93 and 96 (containment isolations) close to isolate the high flow header.

D. CSP-V-97 and 98 (containment isolations) close to isolate the high flow header

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295010AK2.04 2.6/2.8 10CFR55.41 & 45 - Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Nitrogen makeup system.

REFERENCE: LO000162 rev. 6 pages 7 & 8

SOURCE: **NEW QUESTION –** SRO T1, GP1, #8 RO T1, GP1, #4
LO: 5159 – State the auto close signals for nitrogen makeup line containment isolation valves.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: B is incorrect because these valves close on low header temperature. C and D are both incorrect because while these valves do close on hi drywell pressure, they are on the low flow header and have no effect on the hi flow header. A is correct.
COMMENTS:

According to PPM 5.0.10 Flowchart Training Manual, which of the following is a possible consequence of rapid injection of water into the RPV during ATWS conditions?

A. Thermal shock to feedwater nozzles.

B. Fuel damage.

C. Rapid temperature decrease exceeding 100°F/hour.

D. Feed pump trip on high RPV level.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295015AK1.03 3.8/3.9 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: reactivity affects.

REFERENCE: PPM 5.0.10 rev. 6 page 71

SOURCE: **BANK QUESTION – DIRECT –** SRO T1, GP1, #11 RO T1, GP1, #5

LO: 8499 – Given a list, identify the statement that describes the plant response to rapid injection of water into the RPV during an ATWS.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 states that rapid injection into the core during ATWS conditions may result in a large increase in reactivity large enough to cause fuel damage. B is the only correct answer.

COMMENTS:

The plant was operating at 86% power when a fire occurred in the main control room. As the CRS you have directed that the plant be scrammed from the control room and a shutdown be performed from the remote shutdown panel. The following conditions exist:

Reactor Pressure = 945 psig
Reactor Level on the Wide Range = -22 inches
Reactor Level on the Narrow Range = 14 inches
Drywell pressure = 1.58 psig
Drywell Temperature = 128° F

Which of the following is the correct procedure to enter under these conditons

A. PPM 5.1.1 RPV Control

B. PPM 5.2.1 Containment Control

C. ABN-CR-EVAC

D. ABN-RPV-LEVEL

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295016AA2.02 4.3/4.4 10CFR5541, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor Water Level

REFERENCE: LO000126 rev. 8 page 4 and LO000210 rev. 5 page 15

SOURCE: **NEW QUESTION –** SRO T1, GP1, #13

LO: 5885 – List the systems and alignments that can be performed from the RSD Panel

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A and B are incorrect because the EOPS are not entered during control room evacuation conditions. S is incorrect because there is no procedure titled ABN-RPV-LEVEL. C is the correct procedure to enter under these conditons.

COMMENTS:

The plant was operating at 100% power when an accident occurred. The following conditions now exist:

RPV Level = -89 inches and stable
RPV Pressure = 997 psig and stable
Wetwell Level = 19 feet and down slow
Wetwell Temperature = 180°F and up slow
Drywell Pressure = 20 psig and down slow
Drywell Temperature = 300°F and down slow.

Which of the following limits has been exceeded?

A. SRV Tailpipe Level Limit - STPLL

B. Heat Capacity Temperature Limit - HCTL

C. Drywell Spray Initiation Limit - DSIL

D. Primary Containment Pressure Limit - PCPL

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295026 2.1.25 2.3/3.1 10CFR55.41, 43.5, &45 – Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data for High Suppression Pool Water Temperature.

REFERENCE: PPM 5.2.1 SRVTPLL, PCPL, DSIL, and HCTL Curves

SOURCE: **NEW QUESTION –** SRO T1, GP1, #18 RO T1, GP2, #12

LO: 8303- Describe the reason for emergency depressurizing the RPV if wetwell temperature and reactor pressure cannot be maintained below the HCTL.

RATING: H3

ATTACHMENT: **YES –** PPM 5.2.1 SRVTPLL, PCPL, DSIL, and HCTL Curves.

JUSTIFICATION: C and D are both incorrect because DSIL has not yet been exceeded. A is incorrect because Drywell temp is less than the design limit and going down. B is correct because the wetwell temperature has exceeded the HCTL.

COMMENTS:

The plant was operating at 99% power when an accident occurred. The following conditions exist:

An electrical fault has caused a lockout on SM-8.
MS-LIS-38A (level 3 input to ADS) has failed upscale.
Reactor level has been at –135 inches for the last 2 minutes.
All other plant equipment has operated as designed.

Assuming no operator actions, which of the following is correct concerning these conditions?

A. ADS auto initiated from the Division 1 logic.

B. ADS auto initiated from the Division 2 logic.

C. ADS will **not** initiate automatically.

D. ADS will **not** initiate manually.

ANSWER: C

QUESTION TYPE: RO/SRO

KA # & KA VALUE: 218000K1.03 3.7/3.8 10CFR5541 & 45 - Knowledge of the physical connections and/or cause- effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Nuclear Boiler Instrumentation.

REFERENCE: PPM 4.601.A3 rev. 13 page 4 LO000186 rev. 9, page 5 and fig. 7

SOURCE: **NEW QUESTION –** SRO T2, GP1, #12 RO T2, GP1, #17

LO: 5071 – State the condition that will automatically initiate ADS.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: Under the conditions given, ADS will not auto initiate. There is not ECCS Div 2 systems and the Div 1 level 3 confirmatory y switch is out of service. However, because of the arrangement of the logic, ADS could be manually initiated by a manual arm and depress initiation. Therefore, C is the only correct answer.
COMMENTS:

The plant was operating at 100% power when a LOCA occurred. The following conditions exist:

SM-7 is out of service due to a lockout on the bus
RPV level = -172 inches on compensated Fuel Zone and down fast
RPV pressure = 203 psig and down fast
RHR-V-42B is closed with the amber "override" light illuminated
RHR B is in the drywell spray mode

All other plant equipment is normal for these conditions.

According to PPM 5.1.1 RPV Control, which of the following is correct?

A. RHR Loop B can remain in Drywell Spray as long as HPCS is injecting into the core.

B. RHR Loop B can remain in Drywell Spray as long as RHR Loop C is injecting into the core.

C. Open RHR-V-42B: RHR-V-16B and RHR-V-17B can remain open.

D. Open RHR-V-42B and close either RHR-V-16B or RHR-V-17B.

ANSWER: D

QUESTION TYPE: RO/SRO
KA # & KA VALUE: 226001A2.20 3.7/4.1 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: LOCA
REFERENCE: LO000198 rev. 10, pages 16, 19, & 20
SOURCE: **BANK QUESTION – MODIFIED - #LO000835 –** SRO T2, GP1, #16 RO T2, GP2, #10
LO: 5781 – List the interlocks and trips for the RHR system.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: With RPV level less than TAF, injection into the core should be maximized. Only D maximizes flow into the core. D is correct.
COMMENTS:

The plant has been operating at 99% power for the past 3 months. The following conditions exist:

FWLC is operating in 3-Element control with RPV Level instrument "A" as the selected level input.
RPV level instrument "B" is failed downscale.

An open circuit then causes the "A" level instrument to fail downscale. Almost immediately thereafter a similar open circuit causes the "C" level instrument to also fail downscale.

Which of the following is the correct procedure to refer to during the first minute of this event?

A. PPM 3.3.1 Reactor Scram

B. PPM 5.1.1 RPV Control

C. 4.603.A8.3-7 RPV LEVEL HIGH/LOW ALERT

D. 4.603.A8.3-7 RFW CONTR SYSTEM TROUBLE

ANSWER: D

QUESTION TYPE: RO/SRO
KA # & KA VALUE: 259002A2.02 3.3/3.4 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of Reactor Feedwater Inputs.
REFERENCE: LO000157 rev. 11 pages 7 & 8 PPM 4.603.A8.1-7 RFW CONTR SYSTEM TROUBLE
SOURCE: **NEW QUESTION –** SRO T2, GP1, #18 RO T2, GP1, #23
LO: 5400 – Predict the expected response of the feedwater level control system in both Single and Three Element control to a failure of the selected level input.
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: Per the FWLC systems text, a failure of ALL FW level inputs to the FWLC system results in no change in the actual RPV level. FWLC maintains the previous feedpump speed. During steady state operation, RPV level should stay at the setpoint for a considerable period of time. The only indication of a problem would be the RFW CONTR SYSTEM TROUBLE annunciator. C is correct.
COMMENTS:
ex02045

Which of the following describes the basis for the requirement to match Recirculation flows at high power levels?

Matched flows ensure…

A. there are no axial flux anomalies at rated conditions.

B. there are no radial flux anomalies at rated conditions.

C. that during a LOCA due to a Recirc piping break, the assumptions of the LOCA analysis are satisfied.

D. that during a LOCA due to a Recirc piping break, the assumptions of the EOP Basis document are satisfied.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 202001 2.2.25 2.5/3.7 10CFR55.43.2 – Knowledge of basis in Technical Specifications concerning the Recirc System.

REFERENCE: TS 3.4.1 and TS 3.4.1 Bases

SOURCE: **NEW QUESTION –** SRO T2, GP2, #5

LO: 6925 – Identify the basis for any LCO.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A and B are incorrect because as long as the reactor is not operating in the area of increased awareness, there is no concern with reactivity oscillations. D is incorrect because the TS basis for 3.4.1 states as long as the loop flows are matched, the assumptions of the LOCA analysis are satisfied. C is correct.

COMMENTS:

The plant has been operating at 98% power for the last 6 months. A transient occurred 1 minute ago causing several actions including a Main Turbine trip. There are 6 SRVs open and reactor pressure is stable at 1102 psig.

Which of the following procedures takes precedence under these conditions?

A. PPM 5.1.1 RPV Control

B. PPM 5.1.2 RPV Control ATWS

C. ABN-CORE

D. ABN-PRESSURE

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295037EA2.06 4.0/4.1 10CFR55.41, 43.5, & 45 - 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

REFERENCE: PPM 5.0.10 rev. 6, pages 17, 103, & 139

SOURCE: **NEW QUESTION –** SRO T1, GP1, #4

LO: 8017

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: From the conditions given, the reactor is at approximately 36% power in an ATWS. This makes A incorrect. Volume 5 procedures take precedence over ABNs so C and D are incorrect. B is correct due to the ATWS condition.

COMMENTS:

The plant was operating at 99% power when a Main Turbine Trip occurred but the reactor did not scram. Direction in the EOPs is given to manually open SRVs until pressure drops to 945 psig.

Which of the following describes the basis for this direction?

A. Assures that all possible energy is directed to the main condenser.

B. Maximizes the amount of steam condensed in the wetwell.

C. Maintains reactor water inventory in the containment.

D. Maintains pressure below the scram setpoint and allows resetting of the scram.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295037EK3.06 3.8/4.1 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Maintaining heat sinks external to the containment.

REFERENCE: PPM 5.0.10 rev 6, page 175

SOURCE: **BANK QUESTION – 2000 NRC EXAM – EX00044 –** SRO T1, GP1, #23 RO T1, GP1, # 11

LO: 8162 – Given a list identify the advantages of reducing reactor pressure to 945 psig if SRVS are cycling.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 says reducing reactor pressure to 945 psig with SRVs minimizes the addition of energy to the containment. D is correct,

COMMENTS:

The plant was operating at 99% power when a LOCA Signal was received. After verifying auto actions, the CRO notes neither LPCS-P-1 nor RHR-P-2A auto started nor do they have breaker indication on P601. Neither pump will start manually with their control switches on P601.

Which ONE of the following is the correct explanation for these conditions?

A loss of….

A. both B1-1 and C1-1 prior to the LOCA signal

B. both B1-1 and C1-1 after the LOCA signal

C. both B1-2 and C1-2 after the LOCA signal

D. both B1-2 and C1-2 prior to the LOCA signal

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295004AA2.01 3.5/3.9 10CFR55.41, 43.5 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of DC power.

REFERENCE: ABN-ELEC-125VDC rev. 1, page 11

SOURCE: **BANK QUESTION – 2000 NRC EXAM – EX00019 –** SRO T1, GP2, #1

LO: 5262 – Relationship of RHR to 125 VDC

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A loss of Div 1 DC prior to a LOCA signal causes a loss of indication on P601 and a failure of the LPCS pump to start with either the control switch or from a LOCA signal. A is correct. D is incorrect because it references Div 2 DC power.

COMMENTS:

The plant is operating at 100% power when a leak in the CAS System causes pressure to go down.

According to ABN-CAS, which of the following is the highest pressures which would require the direction to verify the opening of CAS-PCV-1(Control Air Desiccant Dryer Bypass)?

A. 72

B. 82

C. 92

D. 102

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295019AA2.01 3.5/3.6 10CFR55.41, 43.5, & 45 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolation valves

REFERENCE: LO000205 rev. 8, page23 ABN-CAS rev. 2 page 2

SOURCE: **NEW QUESTION –** SRO T1, GP2, #7

LO: 5878 – List the expected automatic control air system response to a leak in the control air system.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: CAS-PCV-1 opens at 75 psig. Pressure at 72 psig is below the setpoint of 75 psig and would require the verification of CAS-PCV-1 opening.

COMMENTS:

Which of the following is the reason an Emergency Depressurization is required if Secondary Containment Radiation exceeds its Max Safe Operating Value in more than one area?

Emergency Depressurization…

A. requires a manual scram, which may not have occurred up to this point.

B. reduces the total energy released to the primary and secondary containment.

C. promptly places the primary system in its lowest possible energy state.

D. rejects heat to the main condenser in preference to the reactor building.

ANSWER: C

QUESTION TYPE: SRO\RO

KA # & KA VALUE: 295032EK3.01 3.5/3.8 10CFR55.42 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Emergency Depressurization

REFERENCE: PPM 5.0.10 rev. 6, page 299

SOURCE: **NEW QUESITON –** SRO T1, GP2, #10 RO T1, GP3, #1

LO: 8459 – Given a list identify the statement that describes the two reasons for emergency depressurizing the RPV if on secondary containment parameter is above Maximum Safe Operation Levels in more than one area and a primary system is discharging reactor coolant into secondary containment .

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because Emergency Depressurization does not require a manual scram. B is incorrect because the total energy released is the same, it is just directed to the wetwell. D is incorrect because the Emergency Depressurization is completed using SRVs and not BPVs. C is the reason stated in PPM 5.0.10 and is the correct answer.

COMMENTS:

ex02069

The plant is operating at 88% power when the following auto actions take place:

SGT starts
CSP/CEP isolates
CN makeups isolate
CR and TSC Emerg. Filtration starts and aligns to remote air intakes
RB Emerg. Room Coolers start
RB Lighting quenches
RB EDR and FDR discharge headers isolate

The plant remains operating at power following the initiations.

Which of the following is correct concerning these initiations?

These initiations were caused by a leak…….

A. in the Recirc Pump Suction line.

B. from a Main Steam Line flow element.

C. from RWCU in the Heat Exchanger Room.

D. of spent resin from a RWCU Demin during regeneration.

ANSWER: C	
QUESTION TYPE:	SRO
KA # & KA VALUE:	295034EA2.02 3.7/4.2 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION Cause of the high radiation areas
REFERENCE:	ABN-FAZ-QC pages 3 & 4
SOURCE:	**BANK QUESTION – MODIFIED – 2000 NRC EXAM EX00022 –** SRO T1, GP2, #13
LO:	6914 – Given plant conditions identify those annunciators and indications that would indicate an FAZ actuation and entry into ABN-FAZ.
RATING:	H4
ATTACHMENT:	NONE
JUSTIFICATION:	A and B are incorrect because they would have scrammed the reactor along with starting the listed equipment. D is incorrect because the RWCU demins are located in the Radwaste building and would not cause a Z signal. C is correct because a leak in the RWCU heat exchanger room would cause spread of contamination throughout the reactor building and result in a Z signal trip.
COMMENTS:	

The plant is in MODE 1 at rated conditions with shift turnover in progress. The oncoming Shift Manager has been notified of the absence of the oncoming Mechanical Maintenance person and Plant Laborer due to an automobile accident.

Based on these conditions, which of the following is correct?

A. Both the Maintenance person and the Plant Laborer must be replaced within 2 hours of the start of the shift.

B. One equipment Operator can be designated as an emergency maintenance person, but the on duty Plant Laborer must remain on duty until relieved.

C. Neither the Maintenance person nor the Plant Laborer from the previous shift can leave the plant until relieved by a qualified employee.

D. One Health Physics person can be designated an emergency maintenance person and the Plant Laborer position can be left unmanned.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 600000 2.1.5 2.3/3.4 10CFR55.41, 43.5 – Ability to locate and use procedures and directives related to shift staffing and activities: Fire related.

REFERENCE: PPM 1.3.1 rev 56, page 44 & 45

SOURCE: **BANK QUESTION – DIRECT – 2000 NRC EXAM –** SRO T1, GP21, #16

LO: 6071 – State the minimum staffing level and crew makeup required, both administrative and legal for any given situation.

RATING: H3

ATTACHMENT: **YES** - PPM 1.3.1 rev 56, page 44 & 45

JUSTIFICATION: Because the Laborer is a member of the Plant Fire brigade, he has to be relieved prior to leaving the plant, and must stay until relieved by a qualified Fire Brigade member. The Maintenance person is not a member of the Fire Brigade and can be replaced by one of the equipment operators on shift for emergency maintenance. B is correct.

COMMENTS:

The plant was operating at 89% power when a transient occurred resulting in a scram. It is now several minutes following the scram and the following conditions exist:

Reactor level = 53 inches and rising.
Reactor pressure = 1057 and steady.
RCIC is on at 600 gpm in auto.
HPCS and Feedwater are both off.

Assuming no further operator actions, which of the following will occur given these conditions?

A. RCIC-V-13 and 45 close.
RCIC-V-13 and 45 must be manually reopened to restart RCIC.

B. RCIC-V-13 and 45 close.
Reactor pressure increases and causes a scram signal.
RCIC-V-13 and 45 open when level is reduced to the initiation setpoint.

C. RCIC trips, RCIC-V-1 closes.
RCIC-V-1must be manually reopen to restart RCIC.

D. RCIC trips, RCIC-V-1 closes.
Reactor pressure increases and causes a scram signal.
RCIC-V-1 opens when level is reduced to the initiation setpoint.

ANSWER: B

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	217000K3.02 3.6/3.6 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Reactor vessel pressure
REFERENCE:	LO000180 rev., 11, pages10-12 & 20, 21 lo000161 rev. 11, page11
SOURCE:	**NEW QUESTION –** SRO T2, GP1, #10 RO T2, GP1, #15
LO:	5722 – list the RCIC system isolation signal and setpoints.
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	RCIC-V-45 and 13 auto close at +54.5 inches. They auto open at –50 inches and initiate RCIC. Due to the design of RCIC, steam supply from the vessel and vessel return in the head spray line, it acts to suppress vessel pressure when in operation. When level reaches level 8 and the valves close, reactor pressure increases. B is correct.
COMMENTS:	

Which of the following design features prevents vertical thermal stratification in the Suppression Pool?

A. SRV discharge quenchers are located in the upper elevations of the Suppression Pool.

B. Drywell floor downcomers are located in the upper elevations of the Suppression Pool.

C. RHR discharge lines are located below the elevation of the suctions lines.

D. RHR discharge lines are located above the elevation of the suctions lines.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295013AK1.01 2.5/2.6 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Pool Stratification.

REFERENCE: Columbia FSAR 6.2.2.3 page 6.2-33 & 3
4 and figure 6.2-32

SOURCE: **NEW QUESTION –** SRO T1, GP1, #9 RO T1, GP2, #7

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: C is incorrect because the FSAR states that the RHR discharge lines are located above the elevation of the suction lines for prevention of stratification. A and B are incorrect because they are both located at a lower elevation in the suppression pool. D is correct as stated in the Columbia Generating Station FSAR.

COMMENTS:

ex02051

The plant is operating at 100% power when conditions inside the Reactor Building cause Reactor Building ventilation radiation level to increase to 18 mr/hr.

Which of the following is correct?

A. Manual action should have been taken to isolate Reactor Building Ventilation and start Standby Gas Treatment prior to exceeding the initiation setpoint to prevent a site boundary release.

B. Manual action should have been taken to isolate Reactor Building Ventilation and start Standby Gas Treatment prior to allow a ground level release from the reactor building and prevent a site boundary release.

C. SGT is verified to be in operation and Reactor Building Ventilation is verified to be isolated to allow a ground level release from the reactor building and prevent a site boundary release.

D. SGT is verified to be in operation and Reactor Building Ventilation is verified to be isolated to prevent a radioactive release to the environment from a system that should have isolated.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295038EA1.06 3.5/3.6 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Plant Ventilation

REFERENCE: PPM 5.0.10 rev. 6, page 291

SOURCE: **NEW QUESTION –** SRO T1, GP1, #16 RO T1, GP2, #10

LO: 8460 – Given a list, identify the statement that describes the purpose to confirming RB HVAC isolation and SGT initiation when and FAZ exists.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because auto actions should have occurred before the indicated rad level was this high (13 mr/hr). C is incorrect because operation of SGT and isolation of RB Ventilation do NOT cause a ground level release. D is correct as defined by PPM 5.0.10

COMMENTS:

ex02052

The plant had been operating at 100% power for several months. A normal controlled shutdown to cold conditions is in progress. Reactor power is approximately 20 on range 8 on all IRMs. A DEH malfunction occurs causing reactor power to oscillate and spikes up to 11%. All plant systems function as designed.

As the CRS, which if the following procedures should the CRO be directed to perform?

A. PPM 5.1.2 RPV Control ATWS

B. PPM 3.3.1 Reactor Scram

C. ABN-Pressure

D. ABN-Power

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295014AA2.01 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: Reactor Power

REFERENCE: LO000138 rev. 7, page 8

SOURCE: **NEW QUESTION –** SRO T1, GP1, #17

LO: 5452

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: As stated in the stem, all plant systems function as designed, therefore when pressure spikes to 11% the reactor scrams and PPM 3.1.1 should be entered. A is incorrect because all rods insert. B and C are incorrect because once the reactor scrams they are not entered.

COMMENTS:

What is the basis for the 135° F Tech Spec LCO for Drywell Temperature?

Maintaining Drywell Temperature less than the LCO value assures that the…

A. external design pressure will not be exceed when starting one loop of RHR Drywell Spray.

B. drywell to wetwell interface will not fail during the blowdown portion of a DBA LOCA.

C. equipment inside the Drywell needed to mitigate the effects of a DBA LOCA will operate under conditions expected for the accident.

D. peak post LOCA drywell temperature does not exceed the design temperature of 290° F.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295028EK1.02 2.9/3.1 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification.

REFERENCE: Tech Spec Bases B3.6.1.4 page B3.6.1.4-1

SOURCE: **NEW QUESTION –** SRO T1, GP2, #9 RO T1, GP2, #13

LO: 5635 – List the value for drywell design temperature.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The Basis for TS 3.6.1.4 states that as long as Drywell temperature is maintained less than the TS LCO of 135° F, that equipment inside the containment needed to mitigate the effects of a DBA LOCA will operate as expected. C is correct. A, B, and D are all misapplications or misstatements of other limits.

COMMENTS:

The plant is in MODE 5 with fuel movement underway. A fuel bundle has been dropped and released enough radioactivity into the reactor coolant and to the 606' elevation of the reactor building to result in Reactor Building Exhaust Plenum high radiation of 22 mr/hr.

Which of the following systems is designed to minimize the leakage to the outside atmosphere during these conditions?

A. Control Room Ventilation

B. Secondary Containment

C. Primary Containment

D. Reactor Building Ventilation.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295033EK1.03 3.9/4.2 10CFR55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Radiation release.

REFERENCE: LO000139 rev. 9, page 3

SOURCE: **BANK QUESTION** – **DIRECT – 98 NRC EXAM –** SRO T1, GP2, #12 RO T1, G2, #116

LO: 6999 – State the purpose of Secondary Containment

RATING: L3 L2

ATTACHMENT: NONE

JUSTIFICATION: B is the correct answer. The purpose of the Sec Cont is to minimize the release of radioactivity under these conditions. The other systems either would not limit the spread of radioactivity to the environment or are isolated.

COMMENTS:

The plant is shutdown with RHR-P-2B running in the Shutdown Cooling Mode.

RRC-P-1A is off with RRC-V-67A (1A discharge) closed.
RRC-P-1B is off with RRC-V-67B (1B discharge) closed.
Reactor level is +65 inches

RRC-V-67A is then inadvertently opened.

Which of the following is correct concerning these conditions?

A. No effect on Shutdown Cooling, the suction for RHR-P-2B is from Recirculation Loop B.

B. No effect on Shutdown Cooling, the discharge from RHR-P-2B for Shutdown Cooling goes directly into the core.

C. Shutdown Cooling has bypassed the core and may cause undetected heating of the vessel if level falls below +65 inches.

D. Shutdown Cooling has bypassed the core and may cause undetected heating of the vessel if level falls below +60 inches.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 205000K6.03 3.1/3.2 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM/MODE: Recirculation system.

REFERENCE: SOP-RHR-SDC rev. 0, page 16 and LO000198 rev 10, page 4

SOURCE: **NEW QUESTION –** SRO T2, GP2, #7 RO T2, GP2, #7
LO: 5774 – Describe the flow path for RHR Shutdown Cooling.
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: As stated in PPM 2.4.2, when the 67A valve is opened, a short circuit pathway is opened. If Reactor level is reduced to less than 60 inches, core circulation cannot be assured and undetected heating is the result. D is correct.
COMMENTS:

The plant is in MODE 4. The following conditions exist:

Reactor level = 72 inches
Coolant temperature = 160°F
RRC-P-1A = Off
RRC-P-1B = Off
RHR-P-2B = On in Shutdown Cooling.

What is the basis for the level limitation under these conditions?

Reactor level at 72 inches ensures water level is …

A. above the "turn around" point , there is flow through the core, and thermal stratification will be prevented

B. below the dryer skirt, there is flow from the steam dryer through the core, and thermal stratification is prevented.

C. below the feedwater spargers which promotes better mixing of incoming feedwater to prevent uneven heating of the reactor pressure vessel.

D. above the jet pumps which promotes mixing from the recirculation loops and prevents uneven heating in the reactor pressure vessel.

ANSWER: A

QUESTION TYPE: SRO
KA # & KA VALUE: 290002 2.4.21 (2) 3.7/4.3 10CFR55.43.5 & 45 – Knowledge of the parameters and logic used to assess the status safety functions including: 2 Core cooling and heat removal – Reactor Vessel internals.
REFERENCE: PPM 3.2.1 rev. 46, page 7 LO000198 rev. 10, page 38
SOURCE: **NEW QUESTION –** SROT2, GP3, #2
LO: NO LO
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: Reactor level at 72 inches ensures water is above the turnaround point and that thermal stratification will be avoided. A is correct. B is incorrect because reactor level is above the dryer skirt at 72 inches. C is incorrect because level is above the feedwater spargers. D is incorrect because mixing from the recirc loops does not promote core flow/circulation.
COMMENTS:

As the Shift Manager, you have been notified that an employee with unescorted access has processed through the Primary Access Point and into the protected area without being observed by a Security Officer. There are no Security Officers in the Primary Access point.

Which of the following is the correct action for these conditions?

A. No action is necessary, the employee has unescorted access and passed through the explosive and metal detectors.

B. Notify the NRC within 14 days.

C. Immediately make a verbal notification to the Security Supervisor and notify the NRC within 1 hour.

D. Immediately make a verbal notification to the Security Supervisor and notify the NRC within 14 days.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.13 2.0/2.9 10CFR55.41, 43.5, &45 – Knowledge of the facility requirements for controlling vital/controlled access.

REFERENCE: PPM 1.10.1 rev 22 pages 6 & 11

SOURCE: **BANK QUESTION – 98 NRC ADMIN EXAM –** SRO T3, #2
WNP02 LER 96-S01-00

LO: 6011 – Given procedures determine reportability for a specific event.

RATING: H3

ATTACHMENT: YES – PPM 1.10.1 pages 6 & 11

JUSTIFICATION: PPM 1.10.1 requires that any condition/event that threatens or lessens the physical security of the plant be reported to the Security Supervisor immediately. Furthermore, the actual event was reported under 10cfr73 app G, which requires a 1 hour notification to the NRC. C is the correct answer.

COMMENTS:

The plant is in MODE 5 with control rod removal underway. Control rod 30-31 has to be uncoupled from above the core.

Which of the following tools is used for this action?

A. Fuel support tool

B. Control rod grapple

C. Control rod guide tube grapple

D. Control rod latch tool

ANSWER: D

QUESTION TYPE:	SRO/RO
KA # & KA VALUE:	2.2.27 2.6/3.5 10CFR55.43 – Knowledge of the refueling process.
REFERENCE:	LO000207 rev. 9, page 15
SOURCE:	**BANK QUESTION #3982-** Slightly Modified – SRO T3, #6
LO:	7701 – State the purpose of the control rod latch tool.
RATING:	L2
ATTACHMENT:	NONE
JUSTIFICATION:	The tool used to unlatch and remove a control rod from above the core is the control rod latch tool. D is correct.
COMMENTS:	Changed question for clarification.

The plant is operating at 99% power with the following equipment out of service:

Battery charger C2-1
CRD-P-1A
TO-P-BOP Main Bearing Oil Pump

Which of the following events will cause the 250 VDC battery discharge rate to go up (discharge faster)?

A. Inadvertent HPCS initiation.

B. Inadvertent Div 1 ECCS initiation.

C. Main Turbine Trip.

D. RFW-P-1A trip.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 263000A1.01 2.5/2.8 10CFR55.41 & 45 – Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical Distribution controls including: Battery charging discharging rate.

REFERENCE: LO000188 rev. 6, pages 28 & 29

SOURCE: **NEW QUESTION –** SRO T2, GP2, #11 RO T2, GP2, #14

LO: 5263 – Given a list of loads important to plant safety, identify its relation to 250 VDC bus S2-1.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because they will cause no drain on the 250 VDC system directly, and no drain for any indirect reasons. The trip of the feedpump causes a runback of recirc, but no increased discharge either directly or indirectly on the 250 VDC system. C is correct because the Turbine Trip with the Bearing oil pump out of service causes the Emergency Bearing Oil Pump to start, which is powered from S2-1, 250 VDC.

COMMENTS:

The plant is in MODE 5 with a Full Core Verification in progress.

Which of the following is considered a core alteration?

A. Withdrawal of one SRM with the control switch from the control room.

B. Withdrawal of a control rod from a cell with no fuel.

C. Movement of an irradiated fuel bundle in the Fuel Pool.

D. Reseating of a fuel bundle in the core with the refuel mast.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.2.34 2.3/3.3 10CFR55.43.6 – Knowledge of the process for determining the internal and external effects on core reactivity.

REFERENCE: PPM 6.3.5 rev. 10, page 3

SOURCE: **NEW QUESITON –** SRO T3, #7

LO: 7699 – For a given refueling operation, determine if the evolution is a Core Alteration.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A , B, and C are all incorrect because they do not meet the Tech Spec definition of a core alteration. D is correct because PPM 6.3.5 specifically states the reseating of a fuel bundle during core verification is a core alteration.

COMMENTS:

Which of the following auto actions is correct when a High High Rad signal for Radwaste Effluent is received during a discharge to the river?

A. Alarm only.

B. FDR-V-187 & 188 (RW Effluent) close.

C. FDR-P-45 (RW Eff. Sample Pump) trips, FDR-V-187 & 188 close.

D. CBD-LCV-1 closes to isolate the discharge to the river.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 272000K1.05 2.8/3.1 10CFR55.41 & 45 - Knowledge of the physical connections and/or cause- effect relationships between RADIATION MONITORING SYSTEM and the following: Radwaste System.

REFERENCE: 82-RSY-0400-T6 rev. 9, page18

SOURCE: **NEW QUESTION –** SRO T2, GP2, #12 RO T2, GP2, #15

LO: 5647 – State the auto actions of Radwaste Effluent upon sensing high radiation.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the effluent valves close. C is incorrect because the sample pump does not trip and the valves auto close. D is incorrect because the action to isolate the discharge to the river occurs upstream of the CW System. B is correct.

COMMENTS:

A RWCU Demineralizer resin transfer is in progress when a leak in the system occurs. WEA-RIS-14 Radwaste Building Exhaust and WEA-RIS-14A Radwaste Building Exhaust Extended Range are both in alarm.

Which of the following is correct for these conditions?

A. No auto action occurs, these rad monitors are alarm only.

B. All resin transfer pumps trip immediately.

C. ROA-FN-1A/1B and WEA-FN-1A/1B trip.

D. ROA-FN-1A and WEA-FN-1A trip.

ANSWER: A

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 268000A1.01 2.7/3.1 10CFR55.41 & 45 - Ability to predict and/or monitor changes in parameters associated with operating the RADWASTE controls including: Radiation level

REFERENCE: 82-RSY-0400-T6 rev. 9, page 41

SOURCE: **NEW QUESTION –** SRO T2, GP3, #4 RO T2, GP3, #4

LO: 5647 – State the auto actions of Radwaste Effluent upon sensing high radiation.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: Neither of the WEA radiation indicating switches have auto actions associated with Hi indications, only alarm. A is the only correct response.

COMMENTS:

The plant was operating at 100% power when a transient occurred. As the Shift Manager you have declared an Alert and have become the Emergency Director.

Which of the following can relieve the Shift Manager as the Emergency Director?

A. The OSC Manager or the EOF Manager

B. The TSC Manager or the EOF Manager.

C. The OSC Manager or the TSC Manager

D. The TSC Manager, OSC Manager, or the EOF Manager.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 2.4.40 2.3/4.0 10CFR55.45 – Knowledge of the SROs responsibility in Emergency Plan Implementation.

REFERENCE: PPM 13.1.1 rev. 31, page 3

SOURCE: **NEW QUESTION –** SRO T3, #14

LO: 6132 – Identify the persons who may assume the position of Emergency Director.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As stated in PPM 13.1.1, only the TSC or the EOF Manager can assume the Emergency Director position. The OSC is not permitted to assume this roll. B is correct.

COMMENTS:

The plant was operating at 99% power when a scram occurred. Four control rods stopped at position 02.

Which of the following is correct concerning these conditions?

Reactor shutdown is…

A. not assured; immediately depress the manual scram pushbuttons and initiate ARI.

B. not assured; immediately confirm reactor power is downscale on the APRMs.

C. assured; immediately depress the manual scram pushbuttons and initiate ARI.

D. assured; immediately confirm reactor power is downscale on the APRMs.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295006AA2.01 4.5/4.6 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Reactor power.

REFERENCE: PPM 3.3.1 REV. 39, page 6 PPM 5.0.10 rev. 6, page 186

SOURCE: **NEW QUESTION –** SRO T1, GP1, #20

LO: 8182 – Given a list, identify the criteria that must be met to ensure that the reactor is shutdown with no boron injected.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: C and D are incorrect because reactor shutdown cannot be assured with 4 control rods at position 02. B is incorrect because confirmation of reactor power on the APRMs is not an immediate scram action. A is correct because shutdown cannot be assured and the scram procedure requires that the manual scram pushbuttons be depressed and ARI be initiated immediately if all rods do not insert.

COMMENTS:

ex02063

The plant was operating at 96% power when a transient occurred resulting in high hydrogen concentration in the containment. You have been directed to perform PPM 5.5.16 Emergency Drywell and Wetwell Purging.

Which parameter is of most concern during the performance of this procedure?

A. Suppression Chamber Temperature

B. Suppression Pool Level

C. Drywell Temperature

D. Drywell Pressure

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 500000EA1.07 3.4/3.3 10CFR55.41 & 45 - Ability to determine and/or interpret the following as they apply to SCRAM: Nitrogen purge system.

REFERENCE: PPM 5.5.16 rev. 6, page 3 and PPM 4.603.A7.6-4 rev. 28 page 65

SOURCE: **NEW QUESTION –** SRO T1, GP1, #26 RO T1, GP1, #12

LO: NO LO

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: During a nitrogen purge, containment pressure is the parameter of concern. Without a concurrent containment vent in progress, containment pressure increases. D is correct.

COMMENTS:

ex02064

The plant is operating at 98% power. Maintenance is performing repair work in the area of the Main Turbine Front Standard. Each person is allowed a maximum dose of 300 mrem for the job.

Which of the following events or actions will help this job meet the ALARA Program?

A. Failure of the Main Oil Pump on the HP end of the Main Turbine.

B. Condenser air in-leakage greater than the capacity of Offgas.

C. Assignment of three additional Maintenance Personnel.

D. Rotate personnel out of the rad area for short periods of time.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295005 2.3.2 2.5/2.9 10CFR55.41, 43.4, & 49 – Knowledge of the ALARA program – Turbine Trip.

REFERENCE: LO000129 rev. 9, page 32 LO000161 rev. 11, page 11

SOURCE: **NEW QUESTION –** SRO T1, GP2, #4

LO: 5568 – List parameters and setpoints that will cause a turbine trip.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Because a maximum allowable dose is given, individual stay time can be increased by reduction of the rad field, increasing distance from the source, or adding shielding. Since this is a job on a particular piece of equipment, the distance from the source is fixed. Additional shielding was not given as a possible alternative, therefore only a reduction in the strength of the rad field is possible. C and D are both incorrect because they do not increase the stay time given the maximum dose. A is incorrect because it does not result in a reduction of the rad field; the main turbine does not trip. B is correct because the leak results in a trip of the main turbine and a reactor scram resulting in a reduction of the rad field.

COMMENTS:

ex02065

The plant was operating at 99% power when a transient occurred. The following conditions now exist:

Drywell Temperature = 274°F
Drywell Pressure = 59 psig
Suppression Chamber Pressure = 54 psig
Suppression Pool Temperature = 72°F
Suppression Pool Level = 31 feet

Which of the following is the cause of these conditions?

A. A LOCA with an SRV tail pipe broken above the suppression pool.

B. A LOCA with a broken Drywell Floor.

C. An ATWS with all Bypass Valves failed closed.

D. An ATWS with a broken Drywell Floor Downcomer.

ANSWER: B

QUESTION TYPE: SRO
KA # & KA VALUE: 295028EA2.05 3.6/3.8 10CFR55.41, 43.5, & 44.5 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/Suppression chamber pressure.
REFERENCE: PPM 5.0.10 rev. 6 pages 91 & 92
SOURCE: **NEW QUESITON –** SRO T1, GP1, #8
LO: 8339 – Given a list recognize the primary containment functions that the Pressure Suppression Curve is designed to protect.
RATING: H4
ATTACHMENT: **YES –** PPM 5.0.10 page 91 (without text)
JUSTIFICATION: The only way to achieve the conditions given is to have a steam discharge into the containment with a bypass of the suppression pool: a loss of the pressure suppression function. The only response that results in this condition is B. B is correct. The other responses do not result in the loss of the pressure suppression function.
COMMENTS:

An accident occurred causing a radioactive release. An evacuation has been advised for Sector 1. Meteorological conditions have changed with the wind now blowing from 135°.

Which of the following is correct for these conditions?

Evacuate…

A. Section 2.

B. Section 3a.

C. Section 3b.

D. Section 4

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295038EK1.03 2.8/3.8 10CFRE55.41 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Meteorological effects on Rad release

REFERENCE: Columbia Classification Notification Form (CNF) #968-24075 r16

SOURCE: **NEW QUESTION –** SRO T1, GP1, #24 RO T1, GP2, #17

LO: 8893 – Identify required PARs fro each Emergency Classification.

RATING: H3

ATTACHMENT: **YES –** CNF Form 968-24075 rev 16

JUSTIFICATION: Wind direction has changed 90° from its original direction. It is now blowing from the 135° to 315° , requiring evacuation of Sector 4. D is correct.

COMMENTS:

The plant was operating at 97% power when a complete loss of offsite power occurred. All plant systems responded as designed. Reactor level is –10 inches and up fast. There have been no operator actions since the LOOP. HPCS-P-2 (HPCS Service Water) then trips.

Which of the following is correct for these conditions?

A. DG-3 is allowed to run indefinitely without HPCS-P-2 in operation.

B. DG-3 is allowed to run until the HPCS pump is no longer needed for level control.

C. Immediately trip DG-3 at the local panel with the Emergency Stop pushbutton.

D. Immediately trip DG-3 from P601 with the Emergency Stop pushbutton.

ANSWER: C

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 209002K6.03 2.5/2.6 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): Component Cooling Water system.

REFERENCE: ABN-SW rev. 2, page 2

SOURCE: **NEW QUESITON –** SRO T2, GP1, #4 RO T2, GP1, #7

LO: 6760 – State the immediate actions associated with ABN-SW.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: With a loss of service water to the HPCS DG, ABN-SW requires that DG3 be tripped with the emergency stop pushbutton at the local panel. C is the only correct action.

COMMENTS:

Which of the following activities require the initiation of an Action Request?

A. Relocation of the control switches for SGT-B-1A1 & 1B1.

B. Replacement of the motor operator (same design) for SGT-V-2A.

C. An electrician performs a step out of sequence during an Electrical Surveillance on SGT-V-3A1.

D. An operator finds a valve out of position during verification of a valve lineup on SGT.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 261000 2.2.14 2.1/3.0 10CFR55.43.3, 45 – Knowledge of the process for making configuration changes - SGT

REFERENCE: SWP-AIT-01 rev. 2, page6 SWP-DES-01 rev. 1, page 7

SOURCE: **NEW QUESTION –** SRO T2, GP1, #15

LO: NO LO

RATING: H3

ATTACHMENT: **YES -** SWP-DES-01 rev. 1, page 7

JUSTIFICATION: A is correct because it is a modification to a system in the power block as defined in SWP-DES-1. B is incorrect because the replacement of a motor operator with the same design motor is a maintenance procedure and not a system modification. C and D are both incorrect because they are covered by surveillance procedures.

COMMENTS:

The plant is operating at 99% when reactor level decreases several inches and stabilizes at a new level lower than the setpoint. The Channel A level instrument is selected.

Which of the following is the cause of these indications?

A. MS-RV-4D on Main Steam Line D fails open.

B. MS-V-160A (#1 BPV) fails open.

C. Steam flow transmitter RFW-DPT-803A fails open instantly.

D. Reactor level transmitter RFW-DPT-4A fails downscale instantly.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 259002A3.03 3.2/3.2 10CFR55.41 & 45 - Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Changes in Main Steam Flow

REFERENCE: LO000157 rev. 11, pages 19 & 20

SOURCE: **NEW QUESTION –** RO T2, GP1, #24

LO: 5400 – Predict the expected response of the Feedwater Level Control System with an SRV failed open.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A failure of instrumentation to FWLC causes an instantaneous shift to single element control. The result of this failure is no indicated change in level. C and D are incorrect. B is incorrect because the BPV is located downstream of the flow restrictors and will have no effect on indicated steam flow. A is correct because the open RV causes indicated steam flow to decrease and a corresponding decrease in feed flow until the level signal increase feed flow and stabilizes level at a new lower level.

COMMENTS:

ex02084

The plant is operating at 99% power with all plant equipment operating normally. Several alarms are received in the control room. Upon investigation, the control room operator finds Standby Gas Treatment in operation controlling reactor building pressure. Reactor Building HVAC has tripped. The reactor continues to operate at 99% power.

Considering these conditions, which of the following occurred and would require entry into ABN-FAZ?

A. Loss of one operating RCC Pump.

B. Loss of one reactor feed pump.

C. Leaking fuel in the spent fuel pool.

D. High reactor building dp.

ANSWER: C

QUESTION TYPE: RO
KA # & KA VALUE: 261000A2.12 3.2/3.4 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High fuel pool ventilation radiation.
REFERENCE: 82-RSY-1000-T6 rev. 8, pages1 & 10 LO000144 rev. 11, page14
SOURCE: **NEW QUESTION –** RO T2, GP1, #26
LO: 5679 – Describe the RB HVAC system response to the LOCA isolation signals - FAZ
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: A is incorrect because the loss of one RCC pump only causes the auto start of the standby pump. This results in no loss of RCC and no containment pressure increase. B is incorrect because the loss of one feed pump causes an RRC runback but no level reduction. C is correct because the hi rad/airborne contamination would enter the RB HVAC system from the suction over the fuel pool and cause a Z signal when the ventilation exhaust exceeded 13 mrem/hr. D is incorrect because while a hi reactor building dp would exist for a time, there is no entry into ABN-FAZ.
COMMENTS:

The plant is operating at 89% power. DG-1 has been started and loaded per the monthly operability surveillance. During the operability run, Drywell pressure increases to 2.02 psig. Five minutes later, a lockout occurs on Breaker 7-1.

According to PPM 4.800.C1.2-1, BKR 7/1 OC LOCKOUT, Which of the following should be performed within 6 minutes?

A. Ensure SM-7 is repowered from the Startup Transformer.

B. Ensure SM-7 is repowered from the Backup Transformer.

C. Trip DG-1 using the emergency trip pushbutton in the control room.

D. Trip DG-1 using the normal control switch in the control room.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 264000A2.09 3.7/4.1 10CFR55.41 & 45 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of AC power.

REFERENCE: LO000200 rev. 8, pages 20 & 21

SOURCE: **BANK QUESTION - MODIFIED–** RO T2, GP1, #28

LO: 7773/7774

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A and B are both incorrect because neither will repower the bus as long as there is a lockout. D is incorrect because the normal control switch will not stop the DG with an initiation signal present. C is correct as directed by the procedure.

COMMENTS:

At which of the following power levels will the change in neutron flux cause the most damage during a control rod drop accident?

A. 5

B. 25

C. 35

D. 100

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201003K3.02 2.8/3.1 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Flux shaping

REFERENCE: LO000137 rev. 11, page3 LO000154 rev.10, pages 2 & 3

SOURCE: **NEW QUESTION –** RO T2, GP2, #1

LO: 7283 – State the purpose of flux shaping and rod sequencing.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The damage caused during a control rod drop accident is caused by the change in flux or reactivity. When power exceeds 20% power, there is no operator error that can cause fuel enthalpy to exceed 280 cal./gram. A is correct.

COMMENTS:

ex00003

The plant is operating at rated power when a fault causes an automatic power reduction to approximately 60% of rated.

Assuming no operator actions were taken, which of the following would result in these conditions?

A. SH-6 trip.

B. SM-1 trip.

C. SM-3 trip

D. SM-8 trip

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 202001A4.01 3.7/3.7 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: Recirculation pumps

REFERENCE: LO000196 rev. 12, page 32

SOURCE: **BANK QUESTION – 2000 NRC EXAM** - RO T2, G2, #5

LO: 5030 – State the power supplies for the Recirc pumps.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is correct because the loss of power to RRC-P-1B would result in a power reduction to approximately 60% power. B and C would cause a full scram due to a loss of suction pressure trip on the feedpumps (with no operator action). D would cause a 1/2 scram but no reduction in power.

COMMENTS:

The plant was operating at 100% power when a scram occurred due to an MSIV isolation.

Which of the following caused this isolation?

A loss of …

A. RPS-A

B. RPS-B

C. IN-2

D. IN-3

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 239001K2.01 3.2/3.3 10CFR55.41 - Knowledge of electrical power supplies to the following: MSIV solenoids

REFERENCE: LO000173 rev. 9, page 13

SOURCE: **NEW QUESTION –** SRO T2, GP2, #12

LO: 7783 – Predict the effect a failure of IN2 will have on MSIVs.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A, B, and C are all incorrect because they only de-energize ½ of the logic design needed to cause a full isolation. IN-2 de-energizes both sub channels of the inboard MSIVs causing all 4 to close and a full scram. C is correct.

COMMENTS:

The plant is operating at 99% power with LPCS-P-1 running in full flow test for a surveillance. A small leak in the drywell causes drywell pressure to increase to the LPCS initiation setpoint.

Which of the following valves responds first?

A. LPCS-V-1 Suction

B. LPCS-V-5 Injection

C. LPCS-FCV-11 Minimum Flow

D. LPCS-V-12 Test Return

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295024EA1.03 4.0/3.9 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: LPCS

REFERENCE: LO000192 rev. 9 pages 4 and 5

SOURCE: **BANK QUESTION – LR00688 –** RO T1, GP1, #7

LO: 5482 – List the automatic system response when LPCS initiates.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because there is no auto signal for V-1. B is incorrect because the injection valve does not open until reactor pressure decreases to less than 470 psig. C is incorrect because the min flow valve does not open until flow decreases. D is correct: as soon as the initiation occurs, the Test Return valve closes.

COMMENTS:

Which of the following is prevented by the performance of the PC Gas leg of PPM 5.2.1 Primary Containment Control?

A. Damage to Standby Gas Treatment from excessive hydrogen concentration.

B. An uncontrolled release of radioactivity to the environment.

C. Damage to drywell equipment from drywell sprays.

D. A failure of the drywell downcomers.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 500000 2.3.11 2.7/3.2 10CFR55.45 – Ability to control radiation release during high containment hydrogen.

REFERENCE: PPM 5.0.10 rev. 6 page 267

SOURCE: **NEW QUESTION –** RO T1, GP1, #13

LO: 8425 – Identify the possible consequence of a deflagration in containment.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 states the reason/basis for the PC Gas control leg of PPM 5.2.1 is to prevent the uncontrolled release of radioactivity to the environment. B is correct.

COMMENTS:

The plant is operating at 99% power. The following conditions exist:

Five cooling towers are in operation.
Six Circ Water fans are running on each tower.
Two Circ Water pumps are running.
Condenser vacuum is going down.

Which of the following is true concerning these conditions?

A. Starting all available cooling tower fans causes main generator output to go down because of an increase in plant auxiliary power.

B. Starting all available cooling tower fans causes main generator output to go up because of an decrease in condensate sub cooling.

C. Starting the third Circ Water Pump causes main condenser vacuum to go down because of an increase in Circ Water flow.

D. Starting the third Circ Water Pump causes main condenser vacuum to go up because of an increase in condensate sub cooling.

ANSWER: D

QUESTION TYPE:	RO
KA # & KA VALUE:	295002AA1.07 3.1/2.9 10CFR55.41 & 45 - Ability to operate and/or monitor the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Condenser Circ Water System
REFERENCE:	BWR Gen. Fun. Thermodynamics, Chap. 4, rev. 3, pages35 & 36
SOURCE:	**NEW QUESTION –** RO T1, GP2, #2
LO:	7393 – Explain vacuum formation in the condenser process.
RATING:	L2
ATTACHMENT:	NONE
JUSTIFICATION:	Vacuum can be increased (back pressure decreased) by increasing the condensate sub cooling. Sub cooling can be increased by increasing CW flow or decreasing CW temperature. D is correct because vacuum increases by increasing CW flow (starting the 3rd pump). C is incorrect because vacuum increases. B is incorrect because starting all available fans does not increase CW flow. A is incorrect because main generator output goes up not down.
COMMENTS:	

The plant was operating at 25% power when an ATWS occurred. You have been directed to insert control rods per PPM 5.5.11 Alternate Control Rod Insertions by the CRS.

Which of the following is correct concerning these conditions?

A. The continuous in pushbutton can be used for control rod insertion.

B. Single notch insertion must be used for control rod insertion.

C. Control rods must be inserted according to the fast shutdown sequence.

D. All control rods must be inserted to position 12 before insertion to position 00.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201002 2.4.6 3.1/4.0 10CFR55.41, 43.5, & 45 – Knowledge of the symptom based EOP mitigation strategies – Reactor Manual control system.

REFERENCE: PPM 5.0.10 rev. 6, page 200

SOURCE: **NEW QESTION –** RO T2, GP1, #1

LO: NO LO

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Per the EOP IMPLEMENTATION POLICY given in PPM 5.0.10, the CONTINUOUS IN pushbutton can be used for control rod insertion directed by PPM 5.5.11. A is correct.

COMMENTS:

ex01065

With reactor level GT TAF, operation of the LPCS Pump is **NOT** allowed with suppression pool level LT the Vortex Limit of the pump.

Which of the following describes the reason for this limit?

A. Loss of NPSH results in pump run out and motor overheating.

B. Loss of NPSH results in a pump trip from low suction pressure.

C. Air entrainment can cause pitting and failure in the spray ring nozzles.

D. Air entrainment could occur and cause system damage during subsequent restarts.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 209001K6.03 3.3/3.4 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: Suppression Pool water level

REFERENCE: PPM 5.0.10 rev. 6 pages 262 & 263

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** RO T2, GP1, #5

LO: 8388 - Given a list, identify the statement that describes a centrifugal pump's response to operation below its Vortex Limit.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because a loss of NPSH does not cause pump run out and overheating. B is incorrect because there is no low suction pressure trip on LPCS. C is incorrect because the spray ring is in a high pressure area. D is correct as stated in PPM 5.0.10.

COMMENTS: This question was used for the 2001 exam but with HPCS as the system in question. The only changes to the question were based on LPCS as the system in question versus HPCS. This is considered a direct bank question.

A plant startup is in progress with the following conditions:

Reactor power is approximately 3%
IRM A = 75/125 R8
IRM B = 39/40 R7
IRM C = 65/125 R8
IRM D = 25/40 R7
IRM E = 59/125 R8
IRM F = 60/125 R8
IRM G = 47/125 R8
IRM H = 35/125 R8

The High Voltage supply for IRM E then fails.

Which of the following is correct concerning these conditions?

A. There is no effect on plant operation.

B. Full scram.

C. Rod out Block.

D. ½ scram on RPS A.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 215003K3.04 3.6/3.6 10CFR55.41, & 45 - Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: Reactor Protection System.

REFERENCE: LO000138 rev. 7, page 13
SOURCE: **NEW QUESTION –** RO T2, GP1, #11
LO:
RATING: 5459 – List the IRM scrams and rod blocks
ATTACHMENT: H3
JUSTIFICATION: NONE
The loss of the HV power supply for IRM E causes an INOP scram signal on RPS A. IRM B already has generated a ½ scram on RPS B, so the combination results in a full scram. B is correct.

The plant was operating at 89% power when a transient occurred. The CRS has directed the CRO to open the seven ADS SRVs by arming and depressing the A and C Logic Channel pushbuttons. When the CRO pushes the pushbuttons, the seven ADS SRVs open immediately. All seven ADS SRVs close immediately upon release of the pushbuttons by the CRO.

Which one of the following is correct concerning these conditions?

A. RHR-P-2A is not running.

B. RHR-P-2C is not running.

C. The Division 2 Inhibit switch is in the INHIBIT position.

D. The Division 1 Inhibit switch is in the INHIBIT position.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 218000A4.02 4.2/4.2 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: ADS initiation logic.

REFERENCE: LO000186 rev. 9, page 4

SOURCE: **BANK QUESTON – 2001 NRC EXAM –** RO T2, GP1, #18

LO: 5073 – State how ADS is manually initiated including permissive.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: With all ADS logic made up (all auto contacts made up and the 105 second timer timed out) and the INHIBIT Switches in inhibit, there is no auto initiation. If all ADS logic is made up and the Arm and Depress logic pushbuttons are pushed with the INHIBIT Switches in inhibit, the valves open. When the pushbutton is released, the valves close. D is correct.

COMMENTS:

The plant is operating at 100% power. Multiple alarms illuminate on FCP-1, 2, and 3. Immediately following these alarms a plant laborer calls and informs you there is smoke billowing out of the RFW-P-1B room.

Which of the following is correct for these conditions?

Immediately…

A. Evacuate all non-emergency personnel from the TG Building.

B. Call OPS 3 to verify the fire.

C. Notify the Hanford FD by use of the pushbutton on FCP-1

D. Notify SCC of the location of the fire.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 259001 2.4.25 2.9/3.4 10CFR55.41 & 45 – Knowledge of Fire Protection procedures – Reactor Feedwater Systems

REFERENCE: ABN-FIRE rev. 3, page 2

SOURCE: **NEW QUESTION –** RO T2, GP1, #22

LO: 6902 – Describe the immediate actions for a fire.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because someone has already verified the existence of a fire. C is incorrect because SCC notifies the Hanford FD. D is incorrect because notification of SCC is an immediate action if the fire is outside of the power block. The immediate actions for ABN-FIRE require that all non-emergency personnel be evacuated from the affected building immediately. A is correct.

COMMENTS:

The plant is operating at 100% power. A loss of power occurs on SM-8 due to a lockout.

Which of the following is correct for this condition?

APRM power is indicated on…

A. all normal power indicators on P603.

B. Division 1 power indications on P603.

C. Division 2 power indications on P603.

D. Graphic Display System on P602.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 215005A4.02 2.8/2.8 10CFR55.41 & 45 - Ability to manually operate and/or monitor in the control room: CRT display indicators

REFERENCE: ABN-ELEC-SM3/SM8 rev. 0, page 19 GDS Design Spec. SGPS9401 rev/ 2, APP A table 1.1, page 1

SOURCE: **NEW QUESTION –** RO T2, GP1, #14

LO: NO LO

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The loss of SM-8 causes a loss of all reactor power indication on P603 and a loss of Div 2 APRMs. The GDS system selects the highest power input it receives from the operable APRMs (Div 1) and displays that value. D is correct.

COMMENTS:

During the last refuel outage, two holes (5 inches by 6 inches) were opened in the floor under H13-P682 in the main control room. The plant was started up and is now operating at 97% power.

Which of the following describes the operational implication of operating the plant in this configuration?

A. The normal Control Room Ventilation system flow balance will be changed and will not be able to cool all control room equipment.

B. The Control Room Deluge Systems will not be effective because of the change in the spray patterns caused by the change in ventilation.

C. The Control Room Emergency Filtration System will not be able to pressurize the control room when required during emergency conditions.

D. The Control Room Emergency Filtration System will not be able to filter the incoming atmosphere due to the bypass flow path created.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 290003K5.02 2.8/2.8 10CFR55.41 & 45 - Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Differential pressure control.

REFERENCE: LO000201 rev. 9, pages 2 & 3 **Columbia LER 94-021**

SOURCE: **NEW QUESTON -** RO T2, GP2, #19

LO: 5220 – State the purpose of the Control Room HVAC system.

RATING: H4

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because there would be no change in the flow balance or cooling for control room equipment. B is incorrect because there are not deluge systems in the control room. D is incorrect because the holes in the floor are downstream from the filters and would have no effect on filtration. C is correct as stated in the LER.

COMMENTS:

The following conditions exist:

Isolated ATWS
Reactor level +19 inches and down slow
Reactor pressure 999 psig and up slow
Suppression pool level 31 feet and up slow
Suppression pool temperature 220°F and going up

Which of the following is correct for the above conditions?

A. Start RCIC for reactor level control.

B. Reduce reactor pressure to 600 psig to allow injection with Condensate Booster Pumps.

C. Immediately Emergency Depressurize the reactor.

D. Start HPCS and reduce suppression pool level.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295025EA2.03 3.9/4.1 10CFR55.41, 43.5, &45 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression Pool Temperature.

REFERENCE: PPM 5.2.1, rev 12 - HCTL

SOURCE: **BANK QUESTION – 98 NRC EXAM -** RO T1, G1, #8

LO: 8302 – Given plant conditions, determine current operating point on the HCTL curve.

RATING: H3

ATTACHMENT: YES - PPM 5.2.1
JUSTIFICATION: A and B are both incorrect because level control is secondary to protecting the containment at this time. D is incorrect because suppression pool level is normal and a level reduction would only hurt at this time.
COMMENTS:

Which of the following describes how low pressure LPCI Injection piping is protected from full reactor pressure?

RHR-V-42A (42B and 42C) are interlocked closed until reactor pressure is less than…

A. 160 psig.

B. 220 psig

C. 320 psig.

D. 470 psig.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 203000K4.02 3.3/3.4 10CFR55.41.7 - Knowledge of RHR/LPCI: INJECTION MODE design feature(s) and/or interlocks which provide for the following: Prevention of piping overpressurization.

REFERENCE: LO000198 rev. 10, page 3

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** T2, GP1, #4

LO: 7728 – Describe the physical connections and/or cause and effect relationships between RHR and the RPV.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The LPCI Injection valves are interlocked closed to prevent overpressurization until reactor pressure is reduced to less than 470 psig. D is correct.

COMMENTS:

The plant was operating at 97% power when a LOCA occurred. The LOCA signal is sealed in and has not been reset. All plant equipment functioned as designed. RHR-P-2A is in operation in Upper Drywell Spray. RHR-P-2B is in operation in Wetwell Spray. A lockout on Bkr 7-1 then causes the Startup Transformer to trip.

Which of the following is correct for these conditions?

A. RHR-P-2A is in operation with power from the Backup transformer. RHR-P-2B is in operation with power from the Backup Transformer

B. RHR-P-2A is in operation with power from DG-1. RHR-P-2B is in operation with power from DG-2.

C. RHR-P-2A is off. RHR-P-2B is in operation with power from the Backup Transformer

D. RHR-P-2A is off. RHR-P-2B is in operation with power from DG-2.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 226001K2.02 2.9/2.9 10CFR55.41.7 - Knowledge of electrical power supplies to the following: Pumps

REFERENCE: LO000182 rev. 12. page 30 LO000198 rev. 10, page 43

SOURCE: **BANK QUESTION – 2001 NRC EXAM –** RO T2, GP2, #11

LO: 5058 – Identify the loads on SM-7.

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: RHR-P-2A is powered from SM-7. With the 86 on Bkr 7-1, neither the DG nor the Backup transformer close onto the bus. RHR-P-2A is not in operation. The loss of TR-S causes an undervoltage on SM-8. Bkr B-8 closes and supplies the bus. RHR-P-2B is in operation. C is correct.

COMMENTS:

ex99097

The plant is in MODE 5 with a full core off load 98% completed. RHR-P-2B is in operation in Fuel Pool Cooling Assist mode. Breaker 8-3 trips due to overcurrent.

Which of the following is correct based on these conditions?

A. DG-2 starts and supplies bus SM-8.

B. Breaker B-8 closes and supplies bus SM-8.

C. The spent fuel pool temperature begins to increase.

D. RHR-P-2C will be started in Fuel Pool Cooling Assist Mode.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 233000K2.02 2.8/2.9 10CFR55.41.7 - Knowledge of electrical power supplies to the following: RHR pumps

REFERENCE: LO000202 page 14

SOURCE: **BANK QUESTION- 99 NRC EXAM** RO T2, G3, #3

LO: 5371- Predict how FPCC responds to a loss of RHR.

RATING: H3

ATTACHMENT: N/A

COMMENTS: A trip of breaker 8-3 on overcurrent causes a lockout and none of the power supplies close onto the bus. The result is a loss of RHR-P-2B and a heatup of the spent fuel pool. RHR-P-2C cannot be lined up for FPC Assist. C is correct.

JUSTIFICATION:

ex98086

The reactor is in MODE 5 with fuel movement underway. After moving a bundle through the “cattle chute” and into the vessel cavity, it is observed that the “ROD BLOCK INTERLOCK #1” light does not illuminate. The “HOIST LOADED” indicator is illuminated. The control room reports no rod block indication was received as required.

Which of the following actions is correct for these conditions?

A. Immediately stop the refuel bridge until the inoperable rod block is corrected.

B. Immediately initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies.

C. The fuel bundle may be moved back to the spent fuel pool, then immediately suspend in-vessel fuel movement.

D. Fuel movement may continue as long as ROD BLOCK INTERLOCK #2 is operable.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.11 3.0/3.8 10CFR55.43.2 &45 – Knowledge of less than 1 hour Tech Spec Action statements for systems.

REFERENCE: TS 3.9.1 and TS Bases 3.9.1

SOURCE: **BANK QUESTION – 98 NRC EXAM –** RO T3, #4

LO: 6926 – State from memory, Tech Spec actions required to be taken in less than 15 minutes.

RATING: H3 H4
ATTACHMENT: NONE
JUSTIFICATION: C is correct because TS Bases specifically allow the movement of the component to a safe condition even though the action is for immediate suspension of in vessel fuel movement. Tech Specs does not allow movement to continue until a second fault occurs.
COMMENTS:

The plant is operating at 98% power. At 1500 Wednesday, RHR-P-2C is declared inoperable due to a motor failure. At 1900 Wednesday, DG-1 and all systems supported by the diesel are declared inop.

Which of the following is correct concerning these conditions?

A. Restore RHR-P-2C to operable status in 7 days from 1500 Wednesday.

B. Restore DG-1 to operable status by 1900 Thursday.

C. Perform SR 3.8.1.1 for OPERABLE offsite circuits by 2000 Wednesday, and restore DG-1 to OPERABLE status by 0700 Saturday.

D. Take action within 1 hour (from 1900 Wednesday) to place the unit in MODE 2 within 7 hours, MODE 3 within 13 hours and MODE 4 within 37 hours.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.2.23 2.6/3.8 10CFR55.43.2 & 45 – Ability to track limiting conditions for operations.

REFERENCE: TS. 3.0.3, 3.5.1

SOURCE: **BANK QUESTION - 2000 NRC EXAM –** RO T3, #7

LO: 9540 – Interpret required Tech Specs from plant conditions.

RATING: H4

ATTACHMENT: YES - TS. 3.0.3, 3.5.1

JUSTIFICATION: When DG-1 and all of its supported systems are declared inop, 3 ECCS Systems are out of service and require entry into TS 3.0.3. D is correct.

COMMENTS: This question replaces EX00098 on the RO Exam as number 67.

ex98022

The reactor was operating at 100% power when an MSIV isolation caused reactor pressure to peak at 1156 psig and a reactor scram approximately 5 minutes ago. HPCS and RCIC automatically started and are restoring reactor level. Reactor level is +10 inches and up slow.

Which of the following level instruments is most “accurate” in this situation?

A. Narrow Range

B. Wide Range

C. Upset Range

D. Shutdown Range

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 295007AA2.03 3.7/3.7 10CFR55.41, 43.5, & 45 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor water level

REFERENCE: LO000126 rev. 8, page 4

SOURCE: **BANK QUESTION – 98 NRC EXAM -** RO T1, G1, #3

LO: 5582 – List the calibration conditions for the wide range.

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Because of the high reactor pressure, both RRC pumps have tripped off. A and C are incorrect because the narrow range and upset ranges are calibrated with core flow. D is incorrect because the shutdown range is calibrated at 0 psig and 212°F coolant temperature.

COMMENTS:

The pressure control leg of PPM 5.1.1 RPV Control directs the operator to open SRVs and reduce reactor pressure if any SRV is cycling.

Which of the following is the basis for this direction?

Manual opening of SRVs…

A. minimizes significant dynamic loads on the RPV and containment structures.

B. prevents reactor water level from going above the feedpump trip setpoint.

C. prevents reactor water level from going below the scram setpoint.

D. reduces stress induced on the Main Condenser.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295007AK3.04 4.0/4.1 10CFR55.41 & 45 - Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Safety relied valve operation.

REFERENCE: PPM 5.0.10 rev. 6, page 128

SOURCE: **NEW QUESTION –** RO T1, GP1, #2

LO: 8053 – List advantages of reducing RPV pressure when an SRV is cycling.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: B and C are both incorrect because manual opening of the SRVs may cause both of these setpoints to be exceeded. D is incorrect because the pressure is not supposed to be reduced below the DEH setpoint. This results in no change in the total steam flow to the condenser. A is correct as stated in PPM 5.0.10

COMMENTS:

Which of the following is bypassed by the Mode Switch?

A. Drywell High pressure

B. RPV High Pressure

C. APRM Neutron Flux High

D. SRM Flux High

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 212000K5.01 2.7/2.9 10CFR55.41 & 45 - Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Bypassing of selected scram signals.

REFERENCE: LO000161 rev. 11, page 11

SOURCE: **NEW QUESTION –** RO T2, GP1, #9

LO: 7682 – Describe the physical connection and/or cause and effect relationship between RPS and Neutron Monitoring.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A, B, and D are never bypassed by the Mode Switch, only C is. C is correct.

COMMENTS:

The plant is in MODE 5 with the refuel bridge loaded and over the core. A failure in the Electronic Load Cell causes the sensed load to decrease and the HOIST LOADED Light to go out.

Which of the following is correct for these conditions?

A. A rod block is generated in the control room.

B. The rod withdraw block indicated in the control room clears.

C. The refuel bridge stops at the location where the failure occurred and can only be moved back into the Spent Fuel Pool.

D. The refuel bridge stops at the location where the failure occurred and cannot be moved until the fault is cleared.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 234000K3.01 2.9/3.9 10CFR55.41 & 45 - Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: RMCS

REFERENCE: LO00207 rev. 9, page 9, 26, 31, and 32 LO000148 rev. 10, page 13

SOURCE: **NEW QUESTION –** RO T2, GP3, #2

LO: 5359 – Refuel interlocks in effect when a control rod is withdrawn and Mode Switch is in REFUEL.

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: C and D are both incorrect because the failure does not cause the bridge to stop. The failure causes the existing rod block to clear. B is correct.

COMMENTS:

Which of the following is a control room operator responsibility?

A. Complete the CNF Form.

B. Notify offsite agencies of the event.

C. Function as the Emergency Director in the absence of the Shift Manager.

D. Inform the Shift Manager if any parameter exceeds the emergency action levels.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.39 3.3/3.1 10CFR55.45 – Knowledge of RO responsibilities in emergency plan implementation.

REFERENCE: PPM 13.1.1 rev. 31, pages 3 & 4

SOURCE: **NEW QUESTION –** RO T3, #10

LO: 6130 – Describe the duties of the CRO during emergencies.

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: 13.1.1 states the CRO has the responsibility to notify the Shift Manager of any condition/parameter that exceeds an EAL. D is correct. A, B, and C are incorrect because they are duties/responsibilities of other control room personnel.

COMMENTS: