The plant was operating at 99% power when a loss of battery charger C1-1 occurred. Battery voltage is 92VDC and going down.

Which of the following is correct concerning this condition?

A. RHR-P-2A and LPCS-P-1 will not be available during a LOCA.

B. RHR-P-2B and RHR-P-2C will not be available during a LOCA.

C. TG-EOP-1 Main Turbine Emergency Oil Pump will not be available if needed.

D. RFT-EOP-1A RFP Turbine Emergency Oil Pump will not be available if needed.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295004 AA1.02 – Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: System necessary to assure safe plant shutdown IMP 3.8

REFERENCE: ABN-ELEC-125VDC rev. 1 page 13, SD000188 rev. 7, pages 31 & 32

SOURCE: **BANK QUESTION – MODIFIED –** T1, GP1

LO: 5262, 5263

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Charger C1-1 feeds the div. 1 battery, B1-1. This is a 125 VDC battery. C and D are incorrect because they are 250VDC loads. B is incorrect because it is a div 2, 125VDC load. A is correct as the loss of div 1, 125VDC will cause the loss of ability to operate div 1 breakers.

COMMENTS: Changed may to will in all four distracters.

A plant startup is in progress. Reactor pressure has just been increased to 950 psig and is stable. A leak in the CAS System has resulted in a reduction of CAS pressure to 90 psig and going down. A complete loss of air is imminent.

Which of the following describes the effect of this loss on the Feedwater System?

A. Feedwater heater level control valves will fail as is.

B. Feedwater heater level control valves will fail closed.

C. RFW-FCV-10A & B will fail as is.

D. RFW-FCV-10A & B will fail closed.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295019 AK2.03 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Reactor feedwater IMP 3.5

REFERENCE: SD000205 rev. 9 page 21 & SD000157 rev. 12 page 19

SOURCE: **NEW QUESTION -**

LO: 7605, 5400

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: On a complete loss of CAS, feedwater heater LCVs fail open. A and B are incorrect. RFW-FCV-10A & 10B fail as is. D is incorrect. C is correct.

COMMENTS:

How does boron injection affect reactor power during ATWS conditions?

A. The magnitude of power oscillations during ATWS conditions is reduced by the initiation of the SLC System as the boron concentration in the core increases.

B. The magnitude of power oscillations during ATWS conditions is reduced by the initiation of the SLC System as inlet subcooling increases.

C. Core power goes down because core void fraction goes down as the concentration of boron in the core goes up.

D. Core power goes down because moderator density goes up as the concentration of boron in the core goes up.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295037EK1.03 – Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Boron affects of reactor power IMP 4.2

REFERENCE: PPM 5.0.10 Flowchart Training Manual rev 8, pages 187 & 188

SOURCE: **NEW QUESTION –** T1, GP1

LO: 8086

RATING: L4

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because an increase in core inlet subcooling causes increases in the occurrences of core power oscillations. C is incorrect because a decrease in void fraction causes power to increase. D is incorrect because an increase in moderator density causes a power increase. A is correct as stated in the reference.

COMMENTS:

Which of the following lists EOP actions that mitigate off-site doses due to release of reactor coolant in the secondary containment?

A. Isolate primary systems leaking into secondary containment.
Operate available Reactor Building Ventilation.
Isolate Standby Gas Treatment System.

B. Isolate primary systems leaking into secondary containment
Shut down the reactor.
Operate available Reactor Building Ventilation.

C. Isolate Standby Gas Treatment System.
Shut down the reactor.
Emergency depressurize the reactor.

D. Isolate primary systems leaking into secondary containment.
Shut down the reactor.
Emergency depressurize the reactor.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295038 EK3.02 - - Knowledge of the reasons for the following responses as they apply to HIGH OFF SITE RELEASE RATE: System isolations IMP 3.9

REFERENCE: PPM 5.0.10 rev. 8 pages 312 - 314

SOURCE: **BANK QUESTION LR00884 – Slightly modified for clarity –** T1, GP1

LO: 8460

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 specifically states the actions in D as methods to mitigate rad release. Neither operating Reactor Building Ventilation nor isolating SGT reduces release of radioactivity from the secondary containment. D is correct.

COMMENTS:

EX05027

The plant was operating at 99% power when a scram occurred due to a LOCA concurrent with a lockout on the Startup Transformer. Drywell temperature is going up rapidly. The CRS has directed you to "Maintain drywell temperature below 135°F with available drywell cooling. Upon investigation, you discover all drywell cooling fans have stopped.

Which of the following conditions caused these fans to trip?

A. Reactor level –68 inches.

B. Drywell pressure +2.05 psig.

C. RB Exhaust Plenum 19 mr/hr.

D. Momentary loss of voltage on their respective buses.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295012 AK2.02 – Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling IMP 3.6

REFERENCE: EWD 23E-001 (typical)

SOURCE: **NEW QUESTION –** T1, GP2

LO: 5639

RATING: H4

ATTACHMENT: NONE

JUSTIFICATION: The FAZ signals do not trip the drywell cooling fans. Due to the arrangement of the control switch contacts, the fan stops on loss of voltage and will not restart until the CS is taken to the START position. D is correct.

COMMENTS: Changed signals to conditions in stem.

The plant was operating at 100% power when an ATWS occurred coincident with a loss of MC-7B. Suppression pool temperature is now 115°F and going up.

Which of the following is the current configuration of the SLC System?

A. SLC-P-1A on, SLC-V-1A (suction) open, SLC-V-1B (suction) closed, SLC-V-4A (squib valve) actuated, and SLC-V-4B not actuated.

B. SLC-P-1B on, SLC-V-1A (suction) closed, SLC-V-1B (suction) open, SLC-V-4A not actuated, and SLC-V-4B (squib valve) actuated.

C. SLC-P-1A on, SLC-V-1A /1B (suction) open, and SLC-V-4A/4B (squib valves) actuated.

D. SLC-P-1B on, SLC-V-1A/1B (suction) open, and SLC-V-4A/4B (squib valves) actuated.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 211000 K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM: AC power IMP 3.2

REFERENCE: SD000172 SLC rev. 10, page 19

SOURCE: **BANK QUESTION LX00530 – Slightly modified –** T2, GP1

LO: 5931

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: MC-7B powers the Div 1 equipment, SLC-P-1A, V-1A and V4A. MC-8B powers Div2 equipment, SLC-P-1B, V-1B and V4B. The loss of MC-7B only allows Div 2 equipment to operate. B is correct.

COMMENTS: Changed rating to H3, revised distracters A and B to include other SLC valves

EX05029

APRM A, C, and E just failed downscale.

Which of the following caused these indications?

A loss of…

A. MC-8A

B. 125 VDC division 1

C. 24 VDC distribution panel DP-SO-A

D. Reactor protection system (RPS) bus A

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 215005 K2.02 – Knowledge of the electrical power supplies to the following: APRM channels IMP 2.6

REFERENCE: SD000149 ARPM rev. 10 page 31

SOURCE: **BANK QUESTION – Slightly modified LO00391 –** T2, GP1

LO: 5095

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The power supply for APRM A, C, & E is RPS A as stated n the systems text. D is correct.

COMMENTS:

EX05030

The plant was operating at 89% power when a transient occurred. The CRS has directed the CRO to open the 7 ADS SRVs by arming and depressing the A and C Logic Channel pushbuttons. When the CRO pushes the pushbuttons, the 7 ADS SRVs open immediately. All 7 ADS SRVs close immediately upon release of the pushbuttons by the CRO.

Which of the following is correct concerning these conditions?

A. The Division 1 Inhibit switch is in the INHIBIT position.

B. The Division 2 Inhibit switch is in the INHIBIT position.

C. RHR-P-2A is not running.

D. RHR-P-2C is not running.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 218000 K5.01 - Knowledge of the operational implications of the following as they apply to ADS SYSTEM: ADS logic operation IMP 3.8

REFERENCE: SD000186 ADS rev. 10 page 4

SOURCE: **BANK QUESTION – LO01235 – 2001 NRC Exam –** T2, GP1

LO: 5073

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. A is correct.

COMMENTS:

EX05031

Which of the following functions would be affected by a trip of BKR S-1 when the plant is in Mode 4, with TR-B and DG-1 out of service?

A. High Pressure Core Spray

B. RHR-C LPCI Injection

C. RHR-A Suppression Pool Spray

D. RHR-B Head Spray

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 230000 K2.02 - Knowledge of the electrical power supplies to the following: Pumps IMP 2.8

REFERENCE: SD000198 RHR rev. 11, pages 5 and 49

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5058

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: BKR S-1 fees SM-7 via SM-1. Suppression pool spray could be performed by RHR-P-2A and could be affected by the loss of S-1. C is correct. High and RHR-C pumps are both powered from a different bus. Head spray is on RHR loop B and would not be affected.

COMMENTS: Changed could to would in stem. Added RHR specific loops to C and D.

Columbia Generating Station is in a refueling outage. A full core offload is in progress.

Select the statement below that describes a condition that would prevent moving the refueling bridge over the core.

A. Placing the reactor mode switch in START/HOT STBY with the main hoist grapple is loaded.

B. A single control rod is withdrawn and the main hoist grapple is partially lowered.

C. Placing the reactor mode switch in START/HOT STBY with a control rod selected.

D. Selecting a single control rod while a fuel bundle is loaded on the main hoist grapple.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 234000 K4.01 - Knowledge of FUEL HANDLING EQUIPMENT design features and/or interlocks which provide for the following: Prevention of core alterations during control rod movement IMP 3.3

REFERENCE: SD0000207

SOURCE: **BANK QUESTION – Modified LO00933 –** T2, GP2

LO: 5359

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the main hoist being loaded is not a prerequisite for the interlock of the mode switch in the START/HOT STBY position. B and D are incorrect because by themselves, would not prevent bridge movement over the core. C is correct.

COMMENTS:

The plant is operating at full power when a leak in the CAS system develops. In response to lowering CAS system pressure, CAS-C-1A starts.

Which of the following is correct?

A. If started from STANDBY, CAS-C-1A will run for 1800 seconds and then stop.

B. CAS-C-1A unloaded light will illuminate for a maximum of 1800 seconds.

C. CAS-C-1A will run for 1800 seconds after CAS pressure returns to normal.

D. CAS-C-1A will run until CAS pressure returns to normal and then stop.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.24 Ability to obtain and interpret station electrical and mechanical drawings. IMP 2.8

REFERENCE: EWD-78E-001

SOURCE: **NEW QUESTION –** T3

LO: 4047

RATING: H3

ATTACHMENT: EWD-78E-001

JUSTIFICATION: In standby, CAS-C-1A will run for 30 minutes after the auto start on low pressure clears. C is correct. A is incorrect because the compressor runs until the low pressure clears plus 1800 seconds. B is incorrect because the unload light will be illuminated for as long as CAS-PS-REG/1A is picked up. D is incorrect because CAS-C-1A runs for 1800 seconds after the pressure returns to normal.

COMMENTS: Changed 10 minutes to 30 minutes in justification statement

In accordance with the precautions and limitations associated with SOP-RHR-SDC, which of the following is an acceptable method of providing adequate core flow and also states the reason for assuring adequate core flow?

A. RPV level GT 60”; to promote natural circulation and limit thermal stratification.

B. One Shutdown Cooling Loop in service with RPV level GT +60”; to ensure a minimum RPV piping temperature of GT 70°F.

C. One RRC Pump in operation; to ensure RPV Head temperature remains GT 80°F.

D. One RRC Pump and One SDC Loop in operation; to ensure RPV metal temperature remains LT 200°F.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.32 Ability to explain and apply system limits and precautions IMP 3.4

REFERENCE: SOP-RHR-SDC P&L 4.20 and 4.21 Pages 7 and 8

SOURCE: **NEW QUESTION –** T3

LO: 6486

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: As stated in the P&Ls , the correct answer is RPV level GT 60” to promote natural circulation and limit thermal stratification. A is correct.

COMMENTS:

EX05035

A valve in a high-high rad area has to be closed to prevent uncovering the core.

Which of the following is the maximum administrative TEDE limit for an individual to complete this task?

A. 5 rem TEDE

B. 10 rem TEDE

C. 15 rem TEDE

D. 20 rem TEDE

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 2.3.2 Knowledge of the facility ALARA program IMP 2.5

REFERENCE: GEN-RPP-07 rev. 3, page 8

SOURCE: **BANK QUESTION – LO00351 –** T3

LO: 6016

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: As stated in the procedure, the maximum administrative dose for equipment or health and safety of the public is 10 rem TEDE. B is correct.

COMMENTS: Changed to a BANK QUESTION – not modified.

The reactor was operating at 98% power when a scram occurred. The CRO noted the following conditions:

Reactor pressure	935 psig, down fast
Reactor level	+ 3 inches, down slow
Bypass Valves	100% open
Generator load	572 MWe, down fast
Turbine throttle valves	Closed
Governor valves	Closed

Which of the following is correct concerning subsequent Bypass Valves response?

The Bypass Valves remain full open...

A. until feedwater returns reactor level to GT +13 inches.

B. until intercept valves close.

C. until generator load is less than 25% with a 3-5 second time delay.

D. then close after a 20 second time delay following the turbine trip.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295005AA2.03 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Turbine valve position IMP 3.1

REFERENCE: SD000129 Main Turbine rev. 9, page 20

SOURCE: **BANK QUESTION – 99 NRC Exam - ex99019** - RO T1, G1

LO: 5562

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: C is correct because the DEH opens the BPVs following a scram until main generator load is less than 25% with a 3-5 second time delay.

COMMENTS: Added until to B.

EX02037

Which of the following is correct concerning the basis for the High Drywell Pressure scram?

A. The allowable value is selected to ensure that, for transients involving MSIV isolations, energy discharged to the containment is at its lowest level.

B. The allowable value is selected to ensure that, for transients involving LOCAs initiation of low pressure ECCS at RPV level 1 will not be required.

C. Minimizes the possibility of exceeding ASME code stresses on the primary containment during transients.

D. Minimizes the possibility of fuel damage and reduces the amount of energy being added to the coolant and containment.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295024 EK3.06 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor Scram IMP 4.0

REFERENCE: TS Bases for 3.3.1.1 RPS Instrumentation

SOURCE: **NEW QUESTION –** T1, GP1

LO: 5949

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: D is correct as stated in the TS basis for the LCO. A, B, and C are all incorrect combinations of other bases for RPS instrumentation.

COMMENTS: Changed rating to H3 from L3.

EX05038

The plant was scrammed and EOPs were entered due to increasing temperature from a leak in the drywell.

To which of the following panels should CRO 3 respond first?

A. P840, Bd. A

B. P800, Bd. C

C. P602

D. P601

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295028 2.4.13 – Knowledge of crew roles and responsibilities during EOP flowchart use IMP 3.3

REFERENCE: PPM 1.3.1 rev. 67, page 35

SOURCE: **NEW QUESTION –** T1, GP1

LO: 6088

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM directs that CRO 3 be directed to P601 during the initial stages of a casualty. D is correct.

COMMENTS:

Columbia Generating Station is starting up following a refueling outage. Reactor power is currently 45%. CRO1 notes that reactor power is slowly going up. Scanning his panel he notices that RRC-P-1A speed is slowly rising.

Which of the following is correct for this condition?

A. Immediately SCRAM the Reactor and then stop RRC-P-1A by opening E-CB-RRA or by depressing the STOP pushbutton.

B. Attempt to stop the speed increase of RRC-P-1A, and if unsuccessful, stop RRC-P-1A by opening E-CB-RRA or by depressing the STOP pushbutton.

C. Immediately stop RRC-P-1A by opening E-CB-RRA or by depressing the STOP pushbutton.

D. Attempt to stop the speed increase of RRC-P-1A, and if unsuccessful, SCRAM the Reactor.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 295014 AA1.02 – Ability to operate and or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Recirculation Flow Control system IMP 3.6

REFERENCE: ABN-POWER rev. 4, page 3

SOURCE: **NEW QUESTION –** T1, GP2

LO: 6747

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The immediate actions for a RRC Flow Control System Failure if speed is increasing for one pumps is to attempt to control the speed and if speed cannot be controlled, stop the affected pump. The procedure also states that the preferred method for stopping the pump is to depress the STOP P/B or open E-CB-RRA. B is correct.

COMMENTS:

The plant was operating at 100% power when a transient occurred. Available drywell sprays were put into service per PPM 5.2.1, Primary Containment Control, due to high containment hydrogen and oxygen levels.

After a period of time of spraying the wetwell and the drywell, the following conditions now exist:

Reactor level is -186 inches and stable
Reactor pressure is 75 psig and down slow
Drywell pressure is 1.6 psig and steady
Drywell temperature is 155°F
Drywell and Wetwell hydrogen is 7%
Drywell and Wetwell oxygen is 0.4%
Suppression pool temperature is 101°F
SM-7 has a lockout due to overcurrent relay actuation

Which of the following is the appropriate use of the identified RHR system?

Utilize…..

A. RHR-P-2B for RPV injection due to low RPV level.

B. RHR-P-2A to cool the suppression pool due to high suppression pool temperature.

C. RHR-P-2B to spray the drywell due to high containment hydrogen and oxygen concentrations.

D. RHR-P-2A to spray the drywell due to high drywell temperature.

ANSWER: A

QUESTION TYPE:	RO
KA # & KA VALUE:	500000 EA1.06 - Ability to operate and or monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Drywell sprays IMP 3.3
REFERENCE:	PPM 5.2.1, PPM 5.1.1
SOURCE:	**NEW QUESTION –** T1, GP2
LO:	
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	Reactor level at –186 inches requires that all available ECCS be used for RPV injection. With SM-7 OOS, that leaves only Div. 2 ECCS. B and D are incorrect because they are Div 1 and C is incorrect because the 2B pump is required for RPV injection. A is correct.
COMMENTS:	Rewrote per NRC comments.

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

Reactor level	-280 inches and up slow
Reactor pressure	14 psig
Suppression pool level	+5 inches and stable
Wetwell pressure	12 psig

All ECCS pumps are running on minimum flow.

Which of the following operator actions will cause suppression pool level to go down and stabilize at a substantially lower level?

Start…

A. RPV Head spray with RHR-P-2A

B. Drywell spray with RHR-P-2B

C. RPV injection with RHR-P-2C

D. Wetwell spray with RHR-P-2B

ANSWER: C

QUESTION TYPE:	RO
KA # & KA VALUE:	203000 A1.05 – Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE controls including: Suppression pool level IMP 3.8
REFERENCE:	SD000198 rev. 11, fig. 1
SOURCE:	**NEW QUESTION –** T2, GP1
LO:	5774
RATING:	H2
ATTACHMENT:	NONE
JUSTIFICATION:	A is incorrect because the head spray line is on RHR-B. B and D are incorrect because these flow paths take suction from the wetwell and put the water right back into the suppression pool. C is correct because the pump takes suction from the SP and refills the core shroud back up to the –210 inch level (2/3 core height), which reduces the level in the SP until water spills out of the core shroud and back into the SP.
COMMENTS:	Changed reactor pressure from 10 to 14 psig and -125 to -280 in stem

A transient occurred that has caused reactor level to be –140 inches for the last 115 seconds. No SRVs are open at this point of the transient. Concurrently with this transient, a loss of SM-8 occurred due to an overcurrent. Reactor pressure is currently 485 psig. All other plant equipment operated as expected.

LPCS-V-5…

A. opens when ADS auto initiates and reduces reactor pressure to less than the setpoint.

B. opens after the ADS DIV 1 manual initiation pushbutton is pushed and reactor pressure is reduced to less than the setpoint.

C. opens after the ADS DIV 2 manual initiation pushbutton is pushed and reactor pressure is reduced to less than the setpoint.

D. cannot be opened because ADS will not reduce reactor pressure with a timer failure.

ANSWER: B

QUESTION TYPE:	RO
KA # & KA VALUE:	209001 K6.11 – Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: ADS IMP 3.6
REFERENCE:	SD000192 LPCS rev. 10, page 5 & SD000186 ADS rev. 10, page 4&5
SOURCE:	**NEW QUESTION –** T2, GP1
LO:	5484b, 5071
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	From the conditions given, Div 2 ECCS pumps are not running and Div 2 ADS logic is not made up. Div 1 ADS will not reduce pressure without manual action because the 105s timer should be timed out, but no valves are open. A and C are incorrect. Since reactor pressure is 485 psig, above the valve permissive at 470 psig, the valve is not currently open. D is incorrect. B is correct because the only way the valves will open under these conditions is to push the Div 1 pushbutton. The valves stay open while the button is depressed.
COMMENTS:	

EX05043

A plant startup is in progress with reactor power on range 6 of the IRMs. Annunciator 603.A7 drop 3-5 IRM MONITORS UPSCALE then illuminates. No RPS actuations have occurred.

Which of the following indications caused this alarm?

IRM-A at…

A. 100/125 of scale.

B. 105/125 of scale.

C. 115/125 of scale.

D. 120/125 of scale.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 215003 A3.02 – Ability to monitor automatic operations of the IRM SYSTEM including: Annunciation and alarm signals IMP 3.3

REFERENCE: PPM 4.603.A7.3-5 rev. 30

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5459

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As per the references, the rod block is at 108/125 of scale. Neither A nor B would cause a rod block. And since the stem states that there have been no RPS actuations, D is incorrect. C is correct.

COMMENTS:

Which of the following concerning operation of the Safety Relief Valves is correct?

A. If gas pressure from the CIA is lost, the seven ADS SRVs have an additional 42-gallon accumulator to allow for one manual actuation against maximum drywell pressure with 0 psig reactor pressure.

B. If gas pressure from the CIA is lost, all eighteen SRVs have an additional 10-gallon accumulator to allow for one manual actuation against maximum drywell pressure with 0 psig reactor pressure.

C. All eighteen SRVs can be opened from panel H13-P601 by operation of the associated control switch which energizes either the "A" or the "B" solenoid pilot valve.

D. Position indication for MS-RV-3D, 5B and 5C at the Alternate Remote Shutdown Panel is driven by Linear Variable Differential Transducers (LVDT) mounted on the stem.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 239002 K4.09 – Knowledge of RELIEF/SAFETY VALVES design features and/or interlocks which provide for the following: Manual opening of the SRV
IMP 3.7

REFERENCE: LO000128 Page 4-9

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5528

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The 10 gallon tank is sized for one SRV actuation with RPV/P at 1000 psig and normal DW pressure. H13-P601 operates the "C" solenoid pilot valves. Position indication at the ARSP is dependent on control switch position not LVDT. A is correct as per Systems Text.

COMMENTS:

At the beginning of the fuel shuffle sequence, Source Range Monitor count rate readings are recorded.

Which of the following indications would require stopping of the fuel bundle insertion, close observation of subcritical multiplication behavior while slowly lowering the bundle, and require concurrence of the Reactivity Manager to resume the fuel shuffle?

A. If any SRM count rate exceeds 350 counts/second.

B. If an unexpected two doublings of the average SRM count rate occurs.

C. If a doubling of any single SRM count rate occurs.

D. If an unexpected two doublings of any single SRM count rate occurs.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 259002 2.2.27 – Knowledge of the refueling process IMP 2.6

REFERENCE: PPM 6.3.2 Rev. 17 page 18 of 32

SOURCE: **NEW QUESTION –** T2, GP1

LO: 7700

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Per PPM 6.3.2, if an unexpected doubling in the average SRM count rate occurs, or two doublings of any single SRM count rate occurs, then bundle insertion is stopped and slowly completed. It takes the Reactivity Managers concurrence to resume the shuffle. D is correct.

COMMENTS:

EX05046

The plant is operating at 100% power when a loss of MC-7A occurs.

What effect will this have on the operation of IN-1?

A. The inverter is powered from the normal DC power supply, S1-2.

B. The inverter is powered from the normal DC power supply, S2-1.

C. The Static Switch auto transfers to the backup AC supply, MC-7B.

D. The Static Switch auto transfers to the backup AC supply, MC-8A.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 262001 K3.04 – Knowledge of the effect that a loss or malfunction of the AC ELECTRICAL DISTRIBUTION will have on the following: UPS IMP 3.1

REFERENCE: SD000194 rev. 9, page 4

SOURCE: **BANK QUESTION – 99 NRC EXAM Ex99060 –** T2, GP1

LO: 5896

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The normal AC source for IN-1 is MC-7A. The parallel source is fed from S2-1, 250 VDC. When the primary AC source is lost, the auctioneering diode allows 250 VDC from S2-1 to power the inverter. B is correct.

COMMENTS:

EX05047

A fault has caused the complete loss of DC Bus S1-1.

Which of the effects would this loss have on plant equipment?

Loss of control power to....

A. Bkr 3-8

B. Bkr 1-7

C. Bkr 8-3

D. Bkr 7-1

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 263000 A4.01 - Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses IMP 3.3

REFERENCE: SD000188 rev. 7, pages 24 & 25

SOURCE: **BANK QUESTION – slightly modified 99 NRC Exam ex99084 –** T2, GP1

LO: 5065

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: Bus S1-1 powers division 1 equipment/breaker indication. Only D is a Division 1 load and is correct.

COMMENTS: Changed rating to L2 from H2.

The plant is operating at 100% power. You have just closed RCC-V-6 with the control switch.

Which of the following loads will **NOT** experience a loss of RCC flow?

A. EDR-HX-2 RB Equipment Drain Heat Exchanger

B. FPC-HX-1A Fuel Pool Cooling Heat Exchangers

C. RWCU-P-1A Motor Coolers

D. RRC-P-1A Reactor Recirculation Pump

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 400000 A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the CCWS controls including: CCW flow rate IMP 2.8

REFERENCE: SD000196 rev. 10, page 18

SOURCE: **NEW QUESTION –** T2, GP1

LO: 7668

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: RCC-V-6 isolates all equipment outside of the drywell from the RCC system and causes all flow to be diverted into the drywell. The only load listed in the drywell Is D. It is correct.

COMMENTS:

Which of the following RWCU components/design features is utilized to maximize plant efficiency?

A. CRD purge flow

B. RWCU pump cooling heat exchangers

C. Regenerative heat exchangers

D. Non-regenerative heat exchangers

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 204000 K4.06 - Knowledge of RWCU design features and/or interlocks which provide for the following: Maximize plant efficiency IMP 2.6

REFERENCE: SD000190 rev. 11, page 5

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5034d

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Specifically, the Regen HXs are designed to increase overall plant efficiency by using return water to cool reactor water prior to the demineralizers. C is correct.

COMMENTS:

EX05050

To which of the following systems does the Rod Worth Minimizer input rod blocks for application of control rod movement restrictions?

A. RMCS, Reactor Manual Control System

B. RPIS, Rod Position Information System

C. RSCS, Rod Sequence Control System

D. RBM, Rod Block Monitor System

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201002 K1.05 - Knowledge of the physical connections and/or cause-effect relationships between RMCS and the following: RWM IMP 3.4

REFERENCE: SD000154 RWM, rev. 11, page 5

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5916

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The RWM inputs control rod blocks to the RDCS/RMCS, which imposes the rod movement restrictions. A is correct. The RWM does not interface with RSCS and RBM. The RPIS inputs to the RWM, not from it.

COMMENTS:

Columbia is starting up following a refueling outage. Reactor power is currently 75%. CRO1 is raising power with recirculation flow at a rate of 10 Mwe per minute. During the past few minutes, the Reactor Operator notes that RPV level is slowly trending downward.

Which of the following would explain this unexpected drop in RPV level?

A. The Master Level M/A Station, RFW-LIC-600, has failed as is.

B. RFW-P-1A Speed Controller, RFW-SC-1A, failed to the MDEM Mode.

C. The Startup Valve M/A Station, RFW-LIC-620, has failed as is.

D. RFW-P-1B Speed Controller, RFW-SC-1B, has failed to the MDVP Mode.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 259001K1.08 - Knowledge of the physical and/or cause-effect relationship between REACTOR FEEDWATER SYSTEM and the following: Reactor water level control IMP 3.6

REFERENCE: SD000157 Pg 7 and 17

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5394 and 5400

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: The stem indicates two RFW pumps are in operation. If B or D were to occur, the opposite RFW pump would compensate for the malfunction thus B and D are incorrect. C is incorrect because at this power level the startup controller is not in service. The Master controller is in service and if it fails as is, as power is raised, RPV level drops. A is the correct answer.

COMMENTS: Question replaced after un-secure email sent indicating KA used for this question.

EX05052

The plant is operating at 100% power. The normal cooling water supply for Fuel Pool Cooling has been lost and is not available.

Under these conditions, which of the following systems can be operated from the control room for fuel pool temperature control?

A. RHR Fuel Pool Cooling Assist

B. Reactor Closed Cooling Water

C. Standby Service Water

D. Plant Service Water

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 233000 A4.05 - Ability to manually operate and/or monitor in the control room: Pool temperature IMP 2.5

REFERENCE: SD000202 FPC rev. 11 page 15

SOURCE: **NEW QUESTION –** T2, GP2

LO: 8931

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: RCC is the normal cooling water and is unavailable. TSW cools RCC, which is unavailable, which makes both B and D incorrect. RHR fuel pool assist needs a spool piece installed in the plant and is incorrect. SW is the backup system operated from the control room. C is correct.

COMMENTS:

The plant is operating at 85% power when several alarms are received on back panels. Upon investigation, you note that the Reactor Building exhaust fan tripped off and Reactor Building pressure increased to approximately 4 inches of H_2O. You then noted that Reactor Building pressure drops and stabilized at 0 inches H_2O.

Which of the following explains these indications?

A. ROA-AD-5 Alternate Relief Damper opened to reduce pressure to 0 inches H_2O.

B. ROA-V-1 & 2 and REA-V-1 & 2 closed for pressure control.

C. The Reactor Building supply fan tripped due to high pressure.

D. The standby Reactor Building exhaust fans started due to high pressure.

ANSWER: C

QUESTION TYPE:	RO
KA # & KA VALUE:	290001 K4.02 - Knowledge of SECONDARY CONTAINMENT design features and/or interlocks which provide for the following: Protection against over pressurization IMP 2.8
REFERENCE:	SD000183 RB HVAC rev. 9, pages 8 & 16
SOURCE:	**NEW QUESTION –** T2, GP2
LO:	5680, 5681, 5677b
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	Both REA and ROA fans trip at 4 inches of water for pressure control. C is correct. A is incorrect because ROA-AD-5 will close by 3.5 inches of water. B is incorrect because these valves do not close on high pressure in the reactor building. D is incorrect because the standby fan auto starts only if the running fan remains on.
COMMENTS:	Changed rating to H3 from L3. Revised stem

EX05054

The plant was operating at 98% power when a Station Blackout occurred. The following conditions exist:

4 control rods failed to insert fully
Reactor Level -172 inches

Which of the following actions is required by Tech Specs?

A. Initiate action within 1 hour to restore level to greater than -129 inches.

B. Within 1 hour, restore reactor level to greater than + 13 inches and insert all insertable control rods.

C. Within 2 hours, restore reactor level greater than –161 inches and insert all insertable control rods.

D. Initiate action within 2 hours to restore level to greater than -129 inches.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.2.22 Knowledge of limiting conditions for operations and safety limits
IMP 3.4

REFERENCE: TS 2.1

SOURCE: **BANK QUESTION – EX00005 –** T3

LO: 6934

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Reactor level at –172 inches is a safety limit violation. C is the correct action reactor water level.

COMMENTS:

EX05116 FIXED TYPO IN REFERENCE

Which of the following statements is correct concerning a loss of Control Room Annunciation?

A. If a total loss of Control Room annunciation exists and has been verified, immediately insert a manual scram.

B. If a total loss of Control Room annunciation exists and has been verified, immediately commence a controlled shutdown.

C. If annunciation is lost to H13-P602 and H13-P603 coincident with a control rod drift, perform the immediate actions associated with ABN-ROD.

D. If a loss of Control Room annunciation occurs on H13-P800, H13-P820 and H13-P840 coincident with the performance of a RCIC surveillance, the surveillance may continue.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.32 Knowledge of operator response to loss of all annunciators IMP 3.3

REFERENCE: ~~AND~~ ABN-ANNUN

SOURCE: **NEW QUESTION –** T3

LO: 5262

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because ABN-ANNUN states that a reactor scram is not required as a response to a total loss of annunciation. B is incorrect because ABN-ANNUN states that a controlled shutdown should be made after consultations between the Shift Manager and Plant Management. C is correct as ABN-ANNUN states to suspend operations not essential to safe plant operation. D is incorrect because ABN-ANNUN states to suspend all surveillance testing.

COMMENTS: Re-written due to KA not matching previous question.

EX05056

The reactor was operating at 100% power when a scram occurred. Following the scram the RO notes the UPSCL NEUT FIRST light is illuminated on P608.

Which of the following caused the scram?

A. Loss of both RPS MG Sets.

B. Loss of containment cooling.

C. Trip of both feed pumps.

D. Main turbine trip.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295006AK2.06 – Knowledge of the interrelations between SCRAM and the following: Reactor Power IMP 4.2

REFERENCE: SD000149 rev/ 10, pages 11 & 12

SOURCE: **BANK QUESTION –** T1, GP1

LO: 5089

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The loss of RPS, containment cooling and both feed pumps cause a scram but no large power spike. Loss of the main turbine causes a larger pressure/power spike, which results in an upscale neutron trip before the thermal trip occurs (due to the time constant associated with the thermal trip).

COMMENTS:

EX05057

The control room has been evacuated to the Remote Shutdown Panel due to a fire. RHR-P-2B was in Shutdown Cooling, but the normal discharge flowpath has been lost.

Which of the following is an acceptable flow path for Alternate Shutdown Cooling?

RHR-P-2B in operation discharging through…

A. RHR-V-42B LPCI Injection.

B. RHR-V-53B SDC return.

C. RHR-V-115 Containment Flooding.

D. RHR-V-23 Head Spray.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295021 2.4.35 – Knowledge of local operator auxiliary operator tasks during emergency operations including system geography and system implications. IMP 3.3

REFERENCE: ABN-CR-EVAC rev. 7, page 55

SOURCE: **NEW QUESTION –** T1, GP1

LO: 5574i

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: ABN-CR-EVAC directs the use of RHR-V-42B if the 53B valve is unavailable for injection. A is correct.

COMMENTS: Changed distracter C from RHR-V-24B Full Flow Test.

EX05058

PPM 5.2.1 Primary Containment Control has been entered due to low suppression pool water level.

Which of the following systems is used to add water to the suppression pool from the Condensate Storage Tanks under these conditions?

A. RHR-B

B. HPCS

C. RHR-C

D. RCIC

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 295030 EA1.06 - Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (make up to the suppression pool) IMP 3.4

REFERENCE: PPM 5.5.23 rev. 4

SOURCE: **NEW QUESTION –** T1, GP1

LO: NO LO

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.5.23 gives direction to use HPCS to fill the suppression pool from the CSTs. B is correct.

COMMENTS: Changed A from LPCS and C from RHR.

The plant is operating at 100% power for the past week. Control Room logs are being taken. The operator notes that reactor pressure is three psig higher than it was the last time logs were taken.

Which of the following could have caused a higher reactor pressure?

A. The in-service DEH pressure controller "CONT A OUTPUT" signal slowly failing low.

B. The in-service DEH pressure controller "CONT A OUTPUT" signal slowly failing high.

C. A short voltage dip on US-PP which resulted in a shift to PRESS SET PT MANUAL.

D. Both DEH pressure "CONT A OUTPUT" and "CONT B OUTPUT" have failed and DEH has shifted to BPV MANUAL.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295007AK3.06 – Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Reactor/turbine pressure regulating system IMP 3.7

REFERENCE: SD000146 DEH, rev. 8, page 9, 39 and dwg 4D.

SOURCE: **NEW QUESTION –** T1, GP2

LO: 5286

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As stated in the reference, the HSS selects the controller with the highest output. The standby controller is biased with 3 psi so as the in-service controller fails low, the HSS will transfer control to the standby controller and reactor pressure will have gone up by the 3 psi bias. A is correct. B would result in a lower reactor pressure and C and D do not result in a RPV pressure change.

COMMENTS:

The plant was operating at 100% power when a transient occurred which resulted in a scram. The following indications now exist:

Narrow Range	0 inches
Wide Range	-148 inches and stable
Fuel Zone	-129 inches and stable
Upset Range	0 inches
Drywell Temperature	165° F and up slow
Reactor Pressure	480 psig and down slow

Which of the following is correct concerning these indications?

The RO should report level is…

A. 0 inches

B. -148 inches

C. -129 inches

D. not able to be determined

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295009 AA2.01 – Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level IMP 4.2

REFERENCE: PPM 5.1.1 rev. 16, caution 1 and RPV Saturation Curve.

SOURCE: **NEW QUESTION –** T1, GP2

LO: 8491

RATING: H3

ATTACHMENT: YES - PPM 5.1.1 rev. 16, caution 1 and RPV Saturation Curve.

JUSTIFICATION: It is given in the stem that indications are stable and conditions are less than the sat curve, which indicates there are no inop instrument issues. The WR, NR, and UR are below the MUL, which makes A, B, & C incorrect. The Fuel Zone is on scale and operable. C is correct.

COMMENTS: Changed conditions to indications in stem.
Changed stem to: The RO should report RPV level is…" from "Actual RPV level is…."

EX05061

The plant was operating at 25% power when a transient caused a scram. The following conditions now exist:

Reactor level	+14 inches
Reactor pressure	129 psig
Drywell pressure	1.62 psig

Which of the following is interlocked closed/prevented from opening?

A. RHR-V-24B Full Flow Test

B. RHR-V-53B SDC Return

C. RHR-V-27B Suppression Pool Spray

D. RHR-V-42B LPCI Injection

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 205000 K1.01 – Knowledge of the physical connections and/or cause-effect relationships between SHUTDOWN COOLING SYSTEM and the following: Reactor pressure IMP 3.6

REFERENCE: SD000198 rev. 11, page 30

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5780

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Of the signals listed, only the reactor pressure signal causes an RHR isolation/interlock. This high-pressure interlock prevents the SDC section of piping from being over pressured. The LPCI piping is also protected from over pressurization, but the setpoint is 470 psig. B is the correct answer.

COMMENTS:

Which of the following systems does RCIC share a common injection line for its return to the RPV?

A. HPCS

B. LPCS

C. RHR-A

D. RHR-B

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 217000 K1.05 - Knowledge of the physical connections and/or cause-effect relationships between RCIC SYSTEM and the following: RHR system
IMP 2.6

REFERENCE: SD000180 RCIC rev. 12, page 32 SD000198 rev. 11, figure 1G

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5774, 5726

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: The RHR-B system injects through the RCIC head spray line and tap is between RCIC-V-13 and RCIC-V-65. D is correct.

COMMENTS:

A normal plant startup is in progress with reactor power at approximately 1000 counts on the source range. SRM-B is bypassed for maintenance. During work on SRM-B, the mode switch for SRM-D is inadvertently placed in Standby position.

What effect does this have on the startup?

A. A ½ scram and an upscale trip on SRM-D.

B. A ½ scram and an Inop trip on SRM-D.

C. A rod out block and an upscale trip on SRM-D.

D. A rod out block and an Inop trip on SRM-D.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 215004 A1.06 - Ability to predict and/or monitor changes in parameters associated with operating the SRM controls including: Lights and alarms IMP 3.1

REFERENCE: SD000132 SRM rev. 10, page 26

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5942

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Placing the mode switch in Standby causes an Inop trip on SRM-D. A and C are incorrect. An Inop trip on an SRM causes a rod block and not a scram. D is correct.

COMMENTS:

The plant is operating at 100% power when a loss of control power to RCC-P-1A occurs.

There will be an increase in….

A. CRD pump lube oil temperature.

B. CRD pump motor temperature.

C. drywell air temperature.

D. drywell (EDR-HX-1) Equipment Drain sump temperature.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 201001 K6.06 - Knowledge of the effect that a loss or malfunction of the following will have on the CRDH SYSTEM: CCW system IMP 2.8

REFERENCE: SD000142 CRD rev. 12, page 37

SOURCE: **NEW QUESTION –**T2, GP2

LO: 5706

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The loss of control power to RCC-P-1A closes RCC-V-6. This closure causes a loss of cooling to loads external to the containment. The loss of cooling to CRD results in the increasing temperature of the lube oil for the CRD Pumps. A is correct, B is incorrect. C and D are both incorrect because these coolers do not lose RCC flow.

COMMENTS:

The plant has been shutdown following a LOCA with significant core damage. Drywell Hydrogen concentration is now 10% and Oxygen concentration is at 7%.

In preparation for spraying the drywell, the CRS has directed the RO to stop the drywell recirculation fans per the EOPs. However, one recirculation fan fails to stop due to a control switch malfunction.

Which of the following may occur as a result?

A. Damage to the operating recirculation fan due to water from drywell spray.

B. Inaccurate Drywell Hydrogen indication due to circulation from the running fan.

C. Damage to the operating recirculation fan due to elevated pressures in containment.

D. Hydrogen combustion in the drywell, ignited by the operating recirculation fan.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 223001 K3.04 - Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on the following: Drywell hydrogen gas concentration IMP 3.3

REFERENCE: PPM 5.0.10 rev 9 pg 290 and 298

SOURCE: **NEW QUESTION –** T2, GP2

LO: 8426

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Per PPM 5.0.10, the drywell recirculation fans are secured to eliminate potential ignition sources. D is correct.

COMMENTS: Rewritten per NRC comment

The plant is operating at 100% power. FP-P-2A is running in support of hydrant testing. Fire main pressure is 140 psi. A loss of power to MC-5N occurs and FP-P-2A trips. Fire main pressure is reading 120 psi and trending down quickly. Shortly there after, the FIRE MAIN PRESSURE LOW annunciator illuminates. The ARP directs the operator to check the status of the fire pumps and start them as required.

Which of the following is correct for these conditions?

A. At 120 psig, FP-P-2B Electric pump will start after a 10 second time delay. If pressure continues to drop, at 110 psig, FP-P-1 Diesel pump starts and FP-P-110 Diesel pump will start after a 30 second time delay.

B. At 100 psig, FP-P-2B Electric pump will start. FP-P-1 Diesel pump and FP-P-110 Diesel pump will start when fire main pressure drops to 90 psig.

C. At 110 psig, FP-P-2B Electric pump and FP-P-1 Diesel pump will start. If pressure continues to drop, FP-P-110 Diesel pump will start at 100 psig.

D. At 110 psig, FP-P-2B Electric pump will start after a 10 second time delay and FP-P-1 diesel pump will start after a 15 second time delay. If pressure drops to 100 psig, FP-P-110 Diesel pump will start after a 30 second time delay.

ANSWER: D

QUESTION TYPE:	RO
KA # & KA VALUE:	286000A2.11 - Ability to predict the impacts of the following on the FIRE PROTECTION SYSTEM and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Pump trips IMP 3.1
REFERENCE:	PPM 2.8.7 Rev. 37 page 6
SOURCE:	**NEW QUESTION –** T2, GP2
LO:	5377
RATING:	L3
ATTACHMENT:	NONE
JUSTIFICATION:	Per PPM 2.8.7 FP-P-2B and FP-P-1 start at 110 psig after a 10 and 15 sec. TD. FP-P-110 starts at 100 psig after a 30 second time delay. D is correct.
COMMENTS:	

EX05067

The reactor is operating at 99% power, with HPCS out of service, when a loss of DP-S1-1A occurs. Shortly thereafter, a loss of feedwater initiates a reactor scram. Reactor level is -63” and up slow.

Which of the following describes an action taken based on these conditions?

A. Verify that IN-2 transferred to the alternate AC source.

B. Go to P628 to operate all SRVs for pressure control.

C. Go to P631 to operate the ADS SRVs for pressure control.

D. Verify that IN-1 transferred to the alternate AC source.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.30 Ability to locate and operate components / including local controls IMP 3.9

REFERENCE: LO000128 MS rev. 8, pages 6 & 7

SOURCE: **BANK QUESTION – 98 NRC Exam EX98110 –** T3

LO: 5262

RATING: H4

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because IN-2 is powered from DP-S1-2. B is incorrect because only ADS SRVs can be operated from P628, not all. D is incorrect because IN-1 is powered from 250 VDC not 125 VDC.

COMMENTS:

EX05068

The reactor is in MODE 5. The following conditions exist:

SRM-A	OOS
SRM-B	2.8 counts per second (S/N 9.3/1)
SRM-C	2.5 counts per second (S/N 10.7/1)
SRM-D	OOS
MODE SWITCH	REFUEL
Reactor Coolant Temperature	108°F
Reactor Head and Internals	Removed

It is desired to start a full core off load from the vessel at this time.

Irradiated fuel movement .....

A. may not start at this time because there are no operable SRMs.

B. may not start at this time because reactor coolant temperature is greater than 100°F.

C. may only start in either the quadrant with the B or C SRM.

D. may start in any quadrant in the reactor vessel.

ANSWER: A

QUESTION TYPE: RO
KA # & KA VALUE: 2.2.30 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area / communications with fuel storage facility / system operated from the control room in support of fueling operations/ and supporting instrumentation IMP 3.5
REFERENCE: TS 3.3.1.2 pages 3.3.1.2-1 through 6
SOURCE: **BANK QUESTION – 99 NRC Exam EX99009 –** T3
LO: 10298
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: The TS requires at leas 2 operable SRMs before core alts can start. In this instance, with B and C LT 3 cps, they are not operable and no fuel movement is allowed. A is the correct answer.
COMMENTS: Changed Attachments to NONE.

EX05069

The plant was operating at 98% power when a LOCA occurred. Following the LOCA, LPCS-P-1 is secured, then LPCS PUMP DISCH PRESS HIGH/LO annunciator illuminates. Reactor pressure is 290 psig and going down.

Which of the following is correct if LPCS-P-1 is restarted under these conditions?

A. LPCS-P-1 starts but trips due to overcurrent from excessive flow.

B. LPCS-P-1 starts but does not inject at this pressure.

C. The discharge piping could break resulting in Reactor Building flooding and a reduction in suppression pool level.

D. The discharge piping could break in containment resulting in a reduction in suppression pool level.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 209001K5.05 K5.05 – Knowledge of the operational implications of the following as they apply to LOW PRESSURE CORE SPRAY SYSTEM: System venting IMP 2.5

REFERENCE: PPM 4.601.A3 drop 5-3 rev. 14

SOURCE: **NEW QUESTION –** T2, GP1

LO: 7447

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: A is incorrect because the excessive flow will not cause an overcurrent. B is incorrect because LPCS is injecting at this pressure. D is incorrect because a break in containment would cause suppression pool level to remain static. C is correct as stated in the procedure.

COMMENTS:

EX05070

The reactor was at 100% power when a loss of feedwater caused an auto start of HPCS and RCIC. HPCS-V-4 (injection valve) has been closed to stop HPCS flow. RCIC is injecting into the RPV at 600 gpm. A high suppression pool level annunciator is received. Suppression pool level is +2 inches and increasing.

Which of the following is the reason for the above indications?

A. HPCS on minimum flow from the CSTs causes suppression pool level to increase.

B. RCIC steam exhaust to the suppression pool causes a temperature increase and a false increasing level indication.

C. HPCS on minimum flow causes air entrainment in the suppression pool and a false indicated level increase.

D. RCIC on minimum flow from the CSTs causes suppression pool level to increase.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 209002 K1.02 - Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM and the following: Suppression pool IMP 3.5

REFERENCE: SD000174 HPCS rev. 10, page 4

SOURCE: **BANK QUESTION – 98 NRC Exam EX98044 –** T2, GP1

LO: 5421

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: B is incorrect because a temperature increase from RCIC would not give a false high level indication. C is incorrect because the level upset from minimum flow would not cause a steadily increasing level. D is incorrect because the min flow valve is closed when RCIC is injecting into the vessel.

COMMENTS:

Columbia Generating Station is operating at 100% power. Due to a rupture in its control oil system, RFW-P-1A trips.

Without any operator action, which of the following explains the resultant plant response?

A. At +13 inches the reactor scrams. RPV level drops to approximately -40 inches and returns to approximately +18 inches. Reactor power drops with a stable -80 second period.

B. At +13 inches the reactor scrams. At -50 inches, HPCS and RCIC start, recover RPV level and cycle between +54 inches and -50 inches. Reactor power drops with a stable -80 second period.

C. At +31.5 inches both RRC pumps run back to 51 Hz. Reactor power decreases to approximately 78% power.

D. At +31.5 inches both RRC pumps run back to 30 Hz. Reactor power decreases to approximately 68% power.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 212000 A4.05 – Ability to manually operate and/or monitor in the control room: Reactor power IMP 4.3

REFERENCE: Simulator; SD000184 Rev. 14 page 21

SOURCE: **NEW QUESTION –** T1, GP1

LO: 7670, 9683

RATING: H4

ATTACHMENT: NONE

JUSTIFICATION: A trip of one RFW pump does NOT result in RPV level dropping to the scram setpoint. RRC-P-1A and 1B runback to 30 Hz following a trip of a RFP and a +31inch level input. Power drops to about 50% and returns to about 68%. D is correct.

COMMENTS:

EX05108

Which of the following abnormal procedures directs arming and depressing the NS4 MSIV Isolation Logic A, B, C, and D pushbuttons?

A. ABN-RPS

B. ABN-CR-EVAC

C. ABN-FAZ

D. ABN-CAS

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 223002 2.4.11 – Knowledge of abnormal condition procedures IMP 3.4

REFERENCE: SD000173 NS4 rev. 10, page 12; ABN-CR-EVAC Rev. 7 page 6

SOURCE: **NEW QUESTION –** T2, GP1

LO: 6889

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Depressing the NS4 MSIV Isolation Logic pushbuttons is one of the immediate operator actions associated with ABN-CR-EVAC. B is correct.

COMMENTS:

HPCS was in a normal standby lineup when annunciator HPCS SUCTION SWITCHOVER CST LEVEL LOW illuminated. CST level is 1 foot 6 inches and going down.

Which of the following is directed by the ARP?

Verify…

A. HPCS-V-15 Suppression Pool Suction auto closed
HPCS-V-1 CST Suction auto closed

B. HPCS-V-15 Suppression Pool Suction auto opened
HPCS-V-1 CST Suction auto opened

C. HPCS-V-15 Suppression Pool Suction auto opened
HPCS-V-1 CST Suction auto closed

D. HPCS-V-15 Suppression Pool Suction auto closed
HPCS-V-1 CST Suction auto opened

ANSWER: C
QUESTION TYPE: RO
KA # & KA VALUE: 209002A2.13 A2.13 - Ability to predict the impacts of the following on the HPCS and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Low condensate storage tank level IMP 3.4
REFERENCE: 4.601.A1 drop 5-6 HPCS SUCTION SWITCHOVER CST LEVEL LOW
SD000174 rev. 10, pages 8-10 & 19
SOURCE: **NEW QUESTION –** T2, GP1
LO: 5429
RATING: H3
ATTACHMENT: NONE
JUSTIFICATION: As stated in the procedure and the interlock section of the systems text, only answer C is correct.
COMMENTS: Removed HPCS-V-10 and HPCS-V-11 from each distractor.

A transient has resulted in a LOCA. Reactor level has been –130 inches for the last 60 seconds. All plant equipment is operating as designed. The CRO pushes the Div. 1 Reactor Vessel Low Level/Timer Seal-In pushbutton.

Which of the following is correct concerning these conditions?

A. Only Div.2 ADS SRVs open in 105 seconds to reduce reactor pressure.

B. Only Div.2 ADS SRVs open in 45 seconds to reduce reactor pressure.

C. All ADS SRVs open in 105 seconds to reduce reactor pressure.

D. All ADS SRVs open in 45 seconds to reduce reactor pressure.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 218000A1.04 - Ability to predict and/or monitor changes in parameters associated with operating the ADS controls including: Reactor Pressure IMP 4.1

REFERENCE: SD000186 ADS rev. 10, page 4

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5073

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The low level/timer seal in pushbutton interrupts the time and restarts the timing sequence to 105 seconds. In this case, however, only the Div. 1 timer is reset – Div. 2 continues. When the Div.2 timer times out in 45 seconds, the Div. 2 solenoids energize and open all 7 ADS valves. D is correct.

COMMENTS:

While moving a spent fuel bundle from the core during a refueling outage, the fuel bundle accidentally strikes the edge of the cattle chute while moving into the spent fuel pool. The refuel bridge phone talker then reports seeing bubbles streaming to the surface from the bundle.

What actions are required to be performed immediately?

A. Stop the fuel movement and evacuate all personnel from the refuel floor.

B. Stop the fuel movement and all personnel go to the RB 606' HP control point for further assistance.

C. Continue the fuel movement until the bundle can be place in the correct location in the spent fuel rack.

D. Move the bundle back into the reactor cavity and lower the fuel bundle into the RPV as far as possible to maximize shielding.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295023AA1.05 - Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Fuel transfer system IMP 2.8

REFERENCE: ABN-FUEL-HAND rev. 2, page 2

SOURCE: **BANK QUESTION – 99 NRC Exam EX99081 –** T1, GP1

LO: 6897

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: Answer A is the only response that matches the immediate actions of the procedure.

COMMENTS:

EX05076

The plant was operating at 100% power, when a DEH malfunction caused reactor pressure to go up to 1050 psig.

Which of the following is correct for this condition?

Reactor power…

A. goes down, RPV level is controlled at a new higher level by feedwater level control.

B. goes down, reactor scram and level is controlled by FWLC setpoint setdown.

C. goes up, feedwater level control returns RPV level to the normal range.

D. goes up, RPV level is controlled at a new lower level by feedwater level control.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295025 EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power IMP 4.2

REFERENCE: SD000161 RPS rev. 12, page 12 BWR GFES Reactor Theory, page 52

SOURCE: **NEW QUESTION –** T1, GP1

LO: 7271

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: When pressure goes up, voids decrease, power goes up and level is returned to the normal band by feedwater. A and B are both incorrect because power goes up, not down. D is incorrect because FWLC returns RPV level to the normal range and not a lower level. C is correct.

COMMENTS:

EX05077

The plant is operating at 100% power when UPS E-IN-1 TRIPPED annunciator illuminates. Local investigation indicates that IN-1 has tripped from over voltage.

The ARP refers you to ABN-ELEC-INV, which directs you to verify which of the following?

A. The static switch has auto transferred to Alternate AC input from MC-7A.

B. The static switch has auto transferred to Alternate AC input from MC-7F.

C. The Kirk Key has been manually transferred to Alternate AC input from MC-8A.

D. The Kirk Key has been manually transferred to Alternate AC input from MC-8F.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 262002A2.02 - Ability to predict the impacts of the following on the UPS and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Over voltage
IMP 2.5

REFERENCE: ABN-ELEC-INV rev. 1, page 2 4.800.C1 drop 6-4, SD000194 UPS rev. 9, pages 3-6 & Fig. 3

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5896

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The alternate power supply for the Kirk Key interlock is from MC-7A so C & D are incorrect. The static switch Alternate AC input is from MC-7F which makes A incorrect. MC-7A is the bypass source. MC-7F is the alternate AC input for IN-1, B is correct.

COMMENTS: Completed justification statement.

The plant was operating at 23% power when annunciator MAIN GEN EXCITER FIELD BKR TRIP illuminated.

Which of the following is correct for this condition?

A. Enter ABN-BKR-FAULT.

B. Enter ABN-GENERATOR.

C. Enter PPM 3.3.1 Reactor scram.

D. Enter PPM 5.1.1 RPV Control.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 262002A2.01 – Ability to predict the impacts of the following on the AC ELECTRICAL DISTRIBUTION and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Turbine/generator trip IMP 3.4

REFERENCE: 4.800.C4 drop 8-2

SOURCE: **NEW QUESTION –** T2, GP1

LO: 5520

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: When the field breaker trips, the main generator and turbine trip. Since reactor power was less than 25%, there is no reactor scram. A, C, and D are incorrect. B is the correct answer.

COMMENTS: Revised A from “The main turbine remains in operation as before”.

Which of the following describes the reason Standby Gas Treatment auto starts on the high radiation signal from Reactor Building Ventilation?

Standby Gas Treatment...

A. recirculates and filters reactor building atmosphere during accident conditions to allow personnel entry.

B. maintains a negative pressure in the reactor building of at least 0.5 inches during accident conditions.

C. limits the release of radioactive material within the guidelines of 10CFR100 during accident conditions.

D. provides controlled air movement from areas of potentially low radiation to areas of potentially high radiation.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295034 EK1.02 – Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Personnel protection IMP 3.8

REFERENCE: SD000144 SGT rev. 12, page 3

SOURCE: **BANK QUESTION – MODIFIED - 99 NRC Exam ex99054** – T1, GP2

LO: 5821

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: By definition, SGT reduces the discharge to the atmosphere to less than the limits of 10CFR100. C is correct.

COMMENTS: Reworded distractors B & D to be credible.

Which of the following requires the operator to have a procedure in hand?

When responding to a transient and he is…

A. taking manual control of the FW level controller to prevent exceeding level 8.

B. performing the immediate actions of PPM 3.3.1 Reactor Scram.

C. starting SW-P-1A following a failure to auto start.

D. starting RHR in suppression pool cooling.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 2.1.1 Knowledge of Conduct of Operations Requirements IMP 3.7

REFERENCE: OI-9 rev. 2, page 29

SOURCE: **NEW QUESTION –** T3

LO: 6060

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As stated in OI-9, A, B, and C are specifically exempted from have a procedure present. Only D would require the use of a procedure and is correct.

COMMENTS: Changed stem from “you are” to “he is”.

The plant is operating at 99% power. RHR-P-2B was in operation for surveillance when a fault caused an overcurrent on RHR-P-2B. The overcurrent caused a lockout on BKR 8-3 and a loss of SM-8.

Which of the following is correct for these conditions?

A. The lockout on BKR 8-3 auto resets after the breaker for RHR-P-2B is racked out.

B. The lockout relay must be manually reset at BKR 8-3 after RHR-P-2B is racked out.

C. SM-8 can be manually repowered after RHR-P-2B is racked out.

D. SM-8 will repower automatically after RHR-P-2B is racked out.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 2.4.48 Ability to interpret control room indications to verify the status and operation of system/and understand how operator actions and directives affect plant and system conditions IMP 3.5

REFERENCE: SD000182 AC, rev. 13, page 20

SOURCE: **BANK QUESTION – 03 NRC Exam EX03030 –** T3

LO: 5049d

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Prior to the reset of an 86 device the fault must be cleared from the bus. The 86 then can be manually reset and the breaker can be close manually. B is correct.

COMMENTS: Changed B from “The lockout relay must be manually reset at BKR 8-3 before the bus can be repowered.”.
Changed ‘as soon as’ in all distractors to ‘after’ to reduce sense of urgency.

EX05109

Columbia is operating at 100% power. Due to a tagging error the N2/5 breaker is manually tripped open.

Which of the following explains the plant response to this tagging error?

A. The reactor scrams at +13 inches. RCIC and HPCS auto start at -50 inches and returns RPV level to +54 inches.

B. RPV water level initially drops to LT +31.5 inches. Both Reactor Recirc pumps runback to 51 Hz. RPV level slowly returns back to normal.

C. The reactor scrams at +13 inches. Feedwater system returns reactor water level back to normal.

D. RPV water level initially goes up but remains less than +54 inches. Feedwater system returns reactor water level back to normal.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295001 AK3.01 – Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reactor water level response IMP 3.4

REFERENCE: Simulator

SOURCE: **NEW QUESTION –** T1, GP1

LO: 5023

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: With the trip of 1 RRC pump, core voiding increases which results in an immediate power reduction and an increase in indicated level. This makes D the only correct answer.

COMMENTS:

EX05083

The plant is in Mode 4 with RHR-P-2B in operation in Shutdown Cooling.

Which of the following would cause the loss of RHR-P-2B?

A. Reactor pressure 115 psig

B. Reactor level +10 inches

C. Drywell pressure 1.96 psig

D. RHR SDC flow equal to 5.3 psid

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: AA1.02 - Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling

REFERENCE: SD000173 NS4, rev. 10, pages 5, 12, &21

SOURCE: **NEW QUESTION –** T1, GP1

LO: 5597

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: Of the signals given, only B causes the loss of RHR-P-2B and the loss of shutdown cooling.

COMMENTS:

The plant is operating at 100% power with DG-1 in operation for a surveillance. A fire alarm has been received in the DG-1 room. Subsequent to this annunciator, the standby fire pumps start to sequence on.

What are the required immediate actions for these conditions?

A. Sound the Alerting Tone for 5 seconds
Announce the location of the fire
Repeat the above two steps

B. Sound the Alerting Tone for 5 seconds
Announce evacuation of the Diesel Generator building
Repeat the above two steps

C. Evacuate the Diesel Generator building
Direct the Fire Brigade response using the ROLM PA
Notify the Hanford Fire Department

D. Send an operator to verify the fire is real
Announce the location
Notify the Hanford Fire Department

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 600000 AA2.03 - Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Fire Alarm

REFERENCE: ABN-FIRE rev. 7, page 2

SOURCE: **NEW QUESTION –** T1, GP1

LO: 6902

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: As stated in the procedure, only the actions in A are the correct immediate actions for the indications given.

COMMENTS: Revised B to include ‘announce the..’.
Changed ‘as’ to ‘has’ in the stem.

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

A. Exceeding a maximum hydrogen concentration of 6%.

B. An uncontrolled release of radioactivity to the environment.

C. Exceeding a maximum oxygen concentration of 5%.

D. A failure of the drywell downcomers.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 2.3.11 Ability to control radiation release IMP 2.7

REFERENCE: PPM 5.0.10 rev. 8, page 277

SOURCE: **BANK QUESTION – MODIFIED - 02 NRC Exam EX02074 –** T3

LO: 8425

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: Revised distractors A and C to be credible.

EX05086

During a Control Room Evacuation, transfer switches are repositioned at the Remote Shutdown Panel to the EMERG position to separate equipment control circuits exposed to the control room fire from Safe Shutdown control signals initiated from…

A. DG-1 Local Control Panel
DIV 1 Switchgear Cubicles

B. DG-2 Local Control Panel
DIV 2 Switchgear Cubicles

C. DG-1 Local Control Panel
DIV 2 Switchgear Cubicles

D. DG-2 Local Control Panel
DIV 1 Switchgear Cubicles

ANSWER: B

QUESTION TYPE: RO
KA # & KA VALUE: 295016 AK2.03 – Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations. IMP 4.0
REFERENCE: ABN-CR-EVAC bases 7.2
SOURCE: **NEW QUESTION**
LO: 5886
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: B is the only choice that includes all of the locations where safe shutdown control signals may be initiated.
COMMENTS:

With the plant operating at full power, an RCC leak inside containment occurs. Due to a failure of the operator being able to vent containment, the reactor is scrammed when drywell pressure reached 1.5 psig. Just after the manual scram, the automatic scram at 1.68 psig drywell pressure happens.

Which of the following automatic actions mitigate the effects of this leak?

A. RCC-V-6 Radwaste/Reactor building supply closes.

B. All RCC pumps trip on low surge tank level.

C. All RCC containment isolation valves close.

D. All RCC drywell cooling fan inlet isolation valves close.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 295018 AA1.03 – Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Affected systems so as to isolate damaged portions. IMP 3.3

REFERENCE: SD000196 RCC rev. 10, pages 8-10

SOURCE: **NEW QUESTION**

LO: 5707

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: At 1.68 psig drywell pressure the RCC containment isolation valves close. A is incorrect because RCC-V-6 closes with LT 2 RCC pumps running. B is incorrect because there is no trip of the RCC pumps on low surge tank level. D is not correct because the isolation valves auto open on a fan start and auto close on a fan trip, but the fans have no automatic start/stop.

COMMENTS:

EX05088

Columbia was operating a full power when a series of events occurred that left the plant in an ATWS condition. PPM 5.1.2 was entered from PPM 5.1.1. Per PPM 5.1.2 Power Leg, Boron injection is required before wetwell temperature exceeds 110°F.

Which of the following is the reason for boron injection before wetwell temperature reaches 110°F?

110°F is………..

A. the wetwell temperature at which Technical Specifications requires a reactor scram.

B. the wetwell temperature where no damage to the RCIC system would occur if operation is required to support boron injection.

C. the highest wetwell temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before drywell pressure exceeds the Primary Containment Pressure Limit.

D. the highest wetwell temperature at which initiation of boron injection will permit injection of the Cold Shutdown Boron Weight of boron before wetwell temperature exceeds Heat Capacity Temperature Limit.

ANSWER: A

QUESTION TYPE: RO
KA # & KA VALUE: 295026 EK3.04 – Knowledge of the reasons for the following as they apply to HIGH SUPPRESSION POOL WATER TEMPERATURE: SLC injection IMP 3.7
REFERENCE: PPM 5.0.10 Pg 189 of 318
SOURCE: **NEW QUESTION**
LO: 8086
RATING: L2
ATTACHMENT: NONE
JUSTIFICATION: BIIT is defined as: the wetwell temperature at which Technical Specifications would require a reactor scram or the highest wetwell temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before wetwell temperature exceeds Heat Capacity Temperature Limit. Only A is correct.
COMMENTS:

A loss of all high pressure feed sources has occurred at Columbia. As RPV level drops, PPM 5.1.1, RPV Control, establishes adequate core cooling at different RPV levels. Per PPM 5.1.1, adequate core cooling can be defined four ways:

Core Submergence
Steam Cooling with Injection
Steam Cooling without injection
Spray Cooling

Which of the following is the reason Spray Cooling can provide adequate core cooling?

A. RPV level is maintained GT –201 inches with HPCS or LPCS flow GT 6000 gpm which maintains clad temperature less than 1500°F.

B. RPV level is maintained GT –201 inches with RHR-A, B, or C combined flow GT 6000 gpm which maintains clad temperature less than 1800°F.

C. RPV level is maintained GT –210 inches with RHR-A, B, or C combined flow GT 6000 gpm which maintains clad temperature less than 2200°F.

D. RPV level is maintained GT –210 inches with HPCS or LPCS flow GT 6000 gpm which maintains clad temperature less than 2200°F.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 295031 EK3.03 – Knowledge of the reason for the following responses as they apply to REACTOR LOW WATER LEVEL: Spray Cooling IMP 4.1

REFERENCE: PPM 5.0.10

SOURCE: **NEW QUESTION**

LO: 8018

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: PPM 5.0.10 defines Spray cooling as GT –210” with HPCS or LPCS flow at GT 6000 gpm. Clad temperature will not exceed 2200°F. D is correct.

COMMENTS:

EX05090

Columbia Generating Station is operating in MODE 1 at 95% power. Due to a fault on the bus, a loss of power to S1-2 occurs.

Which of the following describes the operational impact of a loss of S1-2 on the diesel generators?

A. DG-1 cannot be started locally or from the control room.

B. DG-2 cannot be started locally or from the control room.

C. DG-3 could be started locally but not from the control room.

D. DG-2 could be started locally but not from the control room.

ANSWER: B

QUESTION TYPE: RO

KA # & KA VALUE: 264000 K1.02 – Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY DGS and the following: DC electrical distribution. IMP 2.9

REFERENCE: SD000200 PG 26

SOURCE: **NEW QUESTION**

LO: 7653

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Per SD000188, DG-2 cannot be started locally or from the control room. B is the correct answer. The loss of S1-2 has no effect on the other diesels.

COMMENTS:

EX05091

Columbia is in the process of a startup following a refueling outage with power being supplied from TR-S. Preparations for starting the first RRC pump, RRC-P-1A, have been completed. The Channel Selector Switch for Channel 1A1 is in the OFF position and the Channel Selector Switch for Channel 1A2 is in the ON position.

The Reactor Operator momentarily depresses the ASD START pushbutton for RRC-P-1A.

Which of the following describes the expected pump start sequence?

A. ASD Channel 1A2 “READY” light immediately illuminates. RRC-P-1A then starts and ramps to approximately 400 RPM which correlates to 15 Hz.

B. RRC-P-1A starts immediately. Pump speed ramps to approximately 150 RPM which correlates to 15 Hz.

C. RRC-P-1A starts immediately. Pump speed ramps to approximately 450 RPM which correlates to 15 Hz.

D. ASD Channel 1A2 “READY” light immediately illuminates. After a five second time delay RRC-P-1A starts and ramps to 450 RPM which correlates to 15 Hz.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 202001 A3.02 – Ability to monitor automatic operation of the RECIRCULATION SYSTEM including: Pump start sequence. IMP 3.1

REFERENCE: SOP-RRC-START

SOURCE: **NEW QUESTION**

LO: 9681

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Per SOP-RRC-START, the “READY’ light should already be illuminated and the RRC pump immediately starts and increases speed to approximately 450 RPM (15 Hz).

COMMENTS:

Due to a primary coolant pressure boundary leak, the CRS, ordered a reactor scram prior to the automatic scram signal being generated. Wetwell pressure has steadily increased and is currently 15 psig. All systems functioned as designed in response to the event.

Which of the following is correct concerning the spraying of containment at this pressure?

A. If drywell sprays are placed on the 'B' RHR loop, indicated flow will be approximately 8500 GPM.

B. If wetwell sprays are placed on the 'B' RHR loop, indicated flow will be approximately 1000 GPM.

C. If a single loop of RHR was utilized for both wetwell and drywell sprays, indicated flow would be approximately 4200 GPM.

D. If drywell sprays are placed on the 'A' RHR loop, indicated flow will be approximately 6500 GPM.

ANSWER: D

QUESTION TYPE: RO

KA # & KA VALUE: 226001 A3.03 – Ability to monitor automatic operations of THE CONTAINMENT SPRAY SYSTEM MODE including: System flow IMP 2.8

REFERENCE: PLANT SIMULATOR

SOURCE: **NEW QUESTION**

LO: 5774, 5777

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: Pump run out is at approximately 8000 GPM. Wetwell spray flow is approximately 500 GPM and Drywell spray flow is approximately 6500 gpm on the Plant Simulator

COMMENTS:

Columbia is operating at 100% power when the on line after filter for the CAS system becomes clogged. Control air pressure starts to drop.

If efforts are unsuccessful at stopping the pressure decrease, which of the following describes the CAS system response and procedural guidance given to mitigate the consequences of this event?

A. The standby CAS air compressor starts at 100 psig. ABN-CAS directs inserting a manual scram if any outboard MSIV starts to close.

B. The Service Air Compressor, SA-C-1, fully loads at 105 psig. ABN-CAS directs that if any inboard MSIV has closed, place its control switch in the CLOSED position.

C. The Control Air Desiccant Dryer Bypass Valve, CAS-PCV-1, closes at 75 psig. ABN-CAS directs that if two or more rods start to drift, manually scram the reactor.

D. The Service Air Header Isolation Valve, SA-PCV-2, opens at 80 psig. ABN-CAS directs that if a complete loss of air is apparent, manually scram the reactor.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 300000A2.01 – Ability to predict the impacts of the following on the INSTRUMENT AIR SYSTEM and based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions IMP 2.9

REFERENCE: ABN-CAS Rev 5 page 2 and 3

SOURCE: **NEW QUESTION**

LO: 5878

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The standby Comp. starts at 100 psig, CAS-PCV-1 opens at 75 psig. The SA Comp. is always fully loaded when running. SA-PCV-2 closes to 80 psig. ABN-CAS directs inserting a scram if any outboard MSIV starts to close

COMMENTS:

With Columbia operating in MODE 1, RHR-P-2B is running to lower suppression pool level. SW-P-1B is operating in support of the evolution. Due to a problem with the feedwater level control system, the CRS orders a reactor scram prior to the +13" automatic scram. When the Main Turbine trips, TR-S develops an overcurrent condition and a lockout is generated. SM-7 and SM-8 are re-energized from the Backup Transformer.

All systems operate as designed to this event.

Which of the following describes the response of the Service Water System to this event?

When power is restored,…

A. SW-P-1B immediately re-starts. SW-V-2B and SW-V-12B remain open.

B. after a 20 second time delay, and after SW-V-2B is fully closed, SW-P-1B re-starts. When SW-P-1B starts, SW-V-2B re-opens.

C. after a 20 second time delay, and after SW-V-2B and SW-V-12B are both fully closed, SW-P-1B re-starts.

D. SW-V-12B fully closes. When SW-V-12B is fully closed, SW-P-1B re-starts and SW-V-12B re-opens.

ANSWER: B

QUESTION TYPE:	RO
KA # & KA VALUE:	295003 2.1.24 – Ability to obtain and interpret station electrical and mechanical drawings IMP 2.8
REFERENCE:	EWD-58E-004
SOURCE:	**NEW QUESTION**
LO:	4046
RATING:	H3
ATTACHMENT:	EWD-58E-004
JUSTIFICATION:	As per the EWD, SW-V-2B has to be closed before SW-P-1B will restart. There is also a 20 second time delay. SW-V-12B strokes closed but stops mid-stroke when SW-P-1B starts and then goes full open. If SW-V-12B were to go full closed, SW-P-1B would trip.
COMMENTS:	

SGT A has been placed in operation for a surveillance.

Which of the following describes the operation of strip heater, SGT-ESH-1A?

A. SGT-TS-1A21/1A41 open at 125° F and allow the heaters to automatically cycle back on at 110° F.

B. SGT-RLY-ESH1A11/1A31 close at 125° F and allow the heaters to automatically cycle back on at 110° F.

C. SGT-TS-1A11/1A31 and SGT-TC-1A1/1A2 cause the heaters to cycle on and off between 90° F and 110° F.

D. SGT-TS-1A11/1A31 cause the heaters to cycle on at 110° F and SGT-TS-1A21/1A41 cause the heater to cycle off at 125° F.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 261000 2.1.24 – Ability to obtain and interpret station electrical and mechanical drawings. SGT System. IMP 2.8

REFERENCE: EWD-39E-004

SOURCE: **NEW QUESTION**

LO: 4047

RATING: H3

ATTACHMENT: EWD-39E-004 and EWD-46E-157

JUSTIFICATION: Per the EWD SGT-TS-1A11, 1A31 open if temp rises above 110°F and de-energize the heater. SGT-TC-1A1/1A1 are closed when the temp is less than 90° F which causes the heaters to cycle off and on between 90° and 110° F. If temperature increases to GT 125° F, SGT-RLY-ESH1A (CR1) de-energizes, which prevents the operation of the heaters until the reset is pushed. Only C is correct.

COMMENTS: Changed ‘place’ to ‘placed’ in the stem.

The pressure input from the NBI system to the Main Steam Relief Valves has been lost.

What effect does this loss of pressure input from the NBI system have on the operation of the SRVs?

A. The Safety and the Relief mode of operation would remain unaffected.

B. Only the Safety mode of operation would be affected.

C. Only the Relief mode of operation would be affected.

D. Both the Safety and Relief mode of operation would be affected.

ANSWER: C

QUESTION TYPE: RO

KA # & KA VALUE: 239002.K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: NBI pressure indication IMP 3.2

REFERENCE: LO000128

SOURCE: **NEW QUESTION**

LO: 5527; 7638j

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: NBI provides pressure input to the 18 SRVs for operation of the SRVs in the RELIEF mode of operation. The SAFETY mode of operation is actuated directly by the force exerted upon the valve spring by reactor pressure as remains unaffected by a loss of the pressure input.

COMMENTS:

The plant was operating at 25% power when annunciator RHR A PUMP ROOM WATER LEVEL HIGH illuminated. PPM 5.5.27 has been performed and the operator reports RHR-P-2A Room water level as 47 inches.

Which of the following is correct for these conditions?

A. RHR-P-2A may not be able to perform its intended safety function if required.

B. RHR-P-2A may not be operated under any conditions.

C. RHR-P-2A may only be started one time under these conditions.

D. RHR-P-2A may only be started when directed by the Shift Manager.

ANSWER: A

QUESTION TYPE: RO

KA # & KA VALUE: 295036 EA2.01 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Operability of components within the affected area IMP 3.0

REFERENCE: PPM 5.3.1 Sec Cont Control, rev.13 PPM 5.0.10 rev. 8, pages 78 & 301

SOURCE: **NEW QUESTION –** T1, GP2

LO: 8040

RATING: H3

ATTACHMENT: YES – Table 25 from PPM 5.3.1

JUSTIFICATION: The level given in the stem is above the Max Safe Operating Level in Table 25. PPM 5.0.10 states the equipment may not be able to perform its intended safety function with water level greater than max safe. A is correct.

COMMENTS: Changed D from “There are no operational implications with water at this time.

The plant was operating at 99% power when a fire occurred in the control room that required an immediate evacuation. The immediate actions were performed and the following conditions now exist:

Reactor Power is 6% and steady
2 SRVs are cycling open and closed
Reactor level is –15 inches and down slow
Drywell pressure is 1.83 psig and up slow
Suppression Pool level is +3 inches and up slow

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-CR-EVAC

D. PPM 5.2.1 Primary Containment Control

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295016AA2.01 – Ability to determine and/or interpret the following a they apply to CONTROL ROOM ABANDONMENT: Reactor Power 55.43.5 IMP 4.2

REFERENCE: PPM 13.1.1 rev. 33, pages 21 & 36 ABN-CR-EVAC rev. 7, pages 6 &7

SOURCE: **MODIFIED QUESTION 2002 exam –** SRO Tier 1 GP 1

LO: 6105

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Conditions indicate an ATWS that would require an entry into both PPM 5.1.1 and 5.1.2. There are also entry conditions for PPM 5.2.1. However, a control room evacuation is required because of the fire, therefore ABN-CR-EVAC takes precedence as stated on the procedure. C is correct.

COMMENTS:

EX05002

The plant was operating at 99% power when a transient occurred which resulted in an offsite release. The Standby Gas Treatment system is in operation. QEDPS indicates the following dose at the site area boundary:

TEDE dose rate	2 mrem/hr
CEDE thyroid dose rate	159 mrem/hr

Which of the following is correct for these conditions?

The release originates from the…

A. turbine building and PPM 5.4.1 Radioactivity Release Control entry is required.

B. turbine building but PPM 5.4.1 Radioactivity Release Control is **not** required.

C. reactor building and PPM 5.4.1 Radioactivity Release Control entry is required.

D. reactor building but PPM 5.4.1 Radioactivity Release Control is **not** required.

ANSWER: A

QUESTION TYPE: SRO
KA # & KA VALUE: 295038 EA2.04 - Ability to determine and/or interpret the following a they apply to HIGH OFF-SITE RELEASE RATE: Source of off-site release 55.43.5 IMP 4.5
REFERENCE: PPM 13.1.1 rev. 33 page 37
SOURCE: **NEW QUESTION –** SRO T1, GP 1
LO: 5821, 8017
RATING: H3
ATTACHMENT: YES - PPM 13.1.1 rev. 33 page 37, and tables 4 & 5
JUSTIFICATION: As stated in the description and purpose of SGT, radioactive iodines are filtered out in the HEPA filters. A high CEDE dose rate indicates that either SGT is not in operation or the release is from a different source. This makes C and D incorrect. The CEDE dose rate is high enough for an ALERT classification and requires an entry into PPM 5.4.1. A is the correct answer.
COMMENTS:

The plant was operating at 90% power with HPCS out of service for a motor replacement when a loss of both SM-1 & SM-2 occurred and a failure of RCIC to start.

Assume no operator actions are taken for this transient.

Which of the following is the most restrictive time notification required for these conditions?

A. PPM 1.10.1 requires a 4 hour report due to a reactor scram.

B. PPM 1.10.1 requires an 8 hour report due to RCIC failure.

C. GIH-9.1.3 requires that the CEO be notified in 1 hour.

D. GIH-9.1.3 requires that the CEO be notified in 4 hours.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295009 2.1.14 – Knowledge of system status criteria which require the notification of plant personnel 55.43.5 IMP 3.3

REFERENCE: PPM 1.10.1 rev. 26, page 10 GIH-9.1.3 rev. 0

SOURCE: **NEW QUESTION –** T1, GP2

LO: 6011

RATING: H4

ATTACHMENT: YES - PPM 1.10.1 rev. 26, page 10, & 11 GIH-9.1.3 rev. 0

JUSTIFICATION: A is incorrect because the scram requires a 4 hour report. B is incorrect because it would require an 8 hour notification. D is incorrect because GIH 9.1.3 requires a 1 hour notification of the CEO on an emergency classification GT UE. C is correct. In this scenario a SAE would be declared due to RPV level less that -161" as both feed pumps trip on low suction and so there is no HP feed in scenario.

COMMENTS:

The plant is operating at 99% power with the LPCS System Operability Test, OSP-LPCS/IST-Q702, in progress. Surveillance test results reveal the following:

LPCS-FCV-11 opens at 679 gpm
LPCS-FCV-11 closes at 1148 gpm

All other plant equipment is operating as required.

Which of the following is correct for these conditions?

LPCS …

A. must be declared inoperable immediately.

B. must be declared inoperable and the minimum flow function must be restored to operable in 7 days.

C. minimum flow function must be declared inoperable immediately.

D. minimum flow function must be restored to operable in 7 days.

ANSWER: D

QUESTION TYPE:	SRO
KA # & KA VALUE:	209001 2.2.24 – Ability to analyze the affect of maintenance activities on LCO status 55.43.2 IMP 3.8
REFERENCE:	TS 3.3.5.1 and bases.
SOURCE:	**NEW QUESITON –** T2, GP 1
LO:	6925
RATING:	H3
ATTACHMENT:	YES – TS 3.3.5.1, Table 3.3.5.1-1 page 1
JUSTIFICATION:	According to table 3.3.5.1-1, the LPCS minimum flow function is out of spec. 3.3.5.1.d gives the direction to restore the channel to operable in 7 days. Since the redundant feature ECCS is operable, the LPCS system does not have to be declared inop. D is the correct answer.
COMMENTS:	

The plant was operating at 99% power when a transient occurred. The operating crew took all immediate actions. The following conditions exist 10 minutes following the transient:

Reactor level	-66 inches and down slow
Reactor pressure	1096 psig and steady
MSIVs	closed
MS-RV-1A & 1B	open
Suppression pool temp	88° F
Drywell pressure	1.58 psig and stable

Which of the following procedures should have been entered to mitigate the transient?

A. PPM 5.1.1 RPV Control and PPM 5.3.1 Secondary Containment Control

B. PPM 5.1.2 RPV Control ATWS and PPM 5.3.1 Secondary Containment Control

C. PPM 5.1.1 RPV Control and PPM 5.2.1 Primary Containment Control

D. PPM 5.1.2 RPV Control ATWS and PPM 5.2.1 Primary Containment Control

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 295015AA2.01 - Ability to determine and/or interpret the following a they apply to INCOMPLETE SCRAM: Reactor power 55.43.5 IMP 4.3

REFERENCE: PPM 5.1.2 RPV Control ATWS and PPM 5.3.1 Secondary Containment Control entry conditions, LO000128 rev. 8, Main Steam System

SOURCE: **NEW QUESTION –** T1, GP 2

LO: 8017

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: The conditions given indicate approximately 12% steam flow 10 minutes following the transient. This is indicative of an ATWS. A and C are incorrect. With reactor level at –66 inches, RB ventilation supply and exhaust fans have tripped which results in an entry into PPM 5.3.1 from hi Sec. Cont. Pressure. There are no entry conditions at this time for PPM 5.2.1. B is correct.

COMMENTS:

During performance of OSP-ELEC-M702 DG-2 Monthly Operability Test, the equipment operator notifies you that annunciator 3.3 on E-CP-DG/CP2, ENG. 1 LUBE OIL LEVEL LOW is illuminated and there is 8 inches of oil above the LOW mark on the Engine 1 lube oil sump dipstick.

Which of the following is correct for these conditions?

A. Declare DG-2 inoperable immediately.

B. Restore the lube oil inventory to GT 165 gallons in the next 48 hours.

C. Restore the lube oil inventory to GT 283 gallons in the next 48 hours.

D. Restore the lube oil inventory to GT 330 gallons in the next 48 hours.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 264000 2.1.33 – Ability to recognize indications for system operating parameters, which are entry-level conditions for technical specifications. 55.43.2 IMP 4.0

REFERENCE: OSP-ELEC-M702 rev. 20, page 13, 4.DG2 drop 3-3 rev. 10, page 13, and TS 3.8.3 and bases

SOURCE: **NEW QUESTION –** T2, GP1

LO: 10305

RATING: H3

ATTACHMENT: YES - 4.DG2 drop 3-3 rev. 10, page 13, and TS 3.8.3

JUSTIFICATION: The indications given put lube oil level between 283 gal and 330 gal. This requires the lube oil be returned to GT 330 gal in 48 hours. D is correct.

COMMENTS:

The plant was operating at 98% power when the following occurred:

SGT started
CSP/CEP isolated
CN makeups isolated
CR and TSC Emerg Filtration starts and aligns to remote air intakes
RB Emerg Room Coolers start
RB Lighting quenches
RB HVAC isolates and fans trip

The plant remains operating at power following the initiations.

Which of the following is correct concerning these conditions?

Enter…

A. PPM 5.1.1 RPV Control and PPM 5.1.2 RPV Control ATWS

B. PPM 5.1.2 RPV Control ATWS and PPM 5.2.1 Primary Containment Control

C. PPM 5.3.1 Secondary Containment Control and ABN-FAZ

D. PPM 5.4.1 Radioactivity Release Control and ABN-FAZ

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.4.7 Knowledge of how the event based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs 55.43.5 IMP 3.8
REFERENCE:	SD000173 NS4 rev. 10 pages 5, 19, 20 & entry conditions for EOPs/ABN-FAZ
SOURCE:	**BANK QUESTION -** T3
LO:	6914
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	A and B are incorrect because there is no indication given that an ATWS occurred. -50” and 1.68 psig in the drywell would cause these conditions, they would also cause a scram. D is incorrect because the signal causing the indications is a high rad in the exhaust plenum, there is no indication of an offsite release. C is correct because the isolation of RB HVAC causes a high pressure entry into PPM 5.3.1 and an ABN-FAZ entry.
COMMENTS:	

The plant was operating at 99% power when a DEH leak resulted in a DEH pressure reduction to 1100 psig.

Which of the following is correct?

A. A reactor scram occurs because, under these conditions, 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. A reactor scram occurs because, under these conditions, control rod insertion initially adds positive reactivity late in core life that must be compensated for by the trip of both recirc pumps.

C. A reactor scram occurs because, under these conditions, 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.

D. The plant continues to operate at 99% power.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits 55.43.2 IMP 3.7

REFERENCE: TS 3.3.4.1 and Bases, RPS Systems Text SD000161 rev. 12, page 12

SOURCE: **BANK QUESTION –** T3

LO: 6925, 5949

RATING: H3

ATTACHMENT: None

JUSTIFICATION: Since the DEH pressure given is less than the RPS scram setpoint, D is incorrect. The correct basis as in TS is stated in A.

COMMENTS:

If a fire has been reported and confirmed at one of the following locations, to which location should the CRS have the Hanford Fire Department respond to and not the Fire Brigade?

A. Main Transformer Yard within the Protected Area

B. ISFSI within the Protected Area

C. Service Water Pump House 1A

D. ASD Building

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 2.4.27 Knowledge of fire in the plant procedure 55.43.2 IMP 3.5

REFERENCE: ABN-FIRE pg 2 and 15

SOURCE: **NEW QUESTION –** T3

LO: 6783

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Per ABN-FIRE, the CRS/Shift Manager directs the Fire Brigade to respond to areas included in answer A, C, and D. If a fire exists in ISFSI in the protected area, the HFD is contacted.

COMMENTS: Re-wrote question per NRC comment that previous was not an SRO type question.

The plant was operating at 23% power when the Main Turbine tripped and a fuel bundle is dropped in the spent fuel pool during loading of a spent fuel container for ISFSI.

The following conditions exist:

Reactor level is 25 inches and stable
SGT has auto initiated
RB HVAC has isolated
Drywell pressure is 1.1 psig
Wetwell level is –1.9 inches
Drywell temperature is 129 °F

Which of the following actions is correct?

Enter…..

A. PPM 5.1.1 RPV Control and PPM 3.3.1 Reactor Scram.

B. PPM 5.1.1 RPV Control and PPM 5.2.1 Primary Containment Control.

C. PPM 3.3.1 Reactor Scram and PPM 5.3.1 Secondary Containment Control.

D. PPM 5.3.1 Secondary Containment Control and immediately evacuate the refuel floor of all personnel.

ANSWER: D

QUESTION TYPE:	SRO
KA # & KA VALUE:	295023 2.4.1 - Knowledge of EOP entry conditions and immediate action steps 55.43.5 IMP 4.6
REFERENCE:	ABN-FUEL-HAND rev. 2, page 2 PPM 5.0.10 rev. 8, page 294
SOURCE:	**NEW QUESITON –** T1, GP1
LO:	6897, 8017
RATING:	H3
ATTACHMENT:	NONE
JUSTIFICATION:	Since the reactor was at 23% power, there was no scram from the turbine trip. There is no entry given for PPM 3.3.1. Therefore, A-C are incorrect. The immediate action for a dropped fuel bundle is to evacuate the refuel floor and 5.3.1 must be entered because of the high exhaust plenum rad as indicated by the auto SGT start. D is correct.
COMMENTS:	

Columbia is in day 15 of a 25 day refueling outage. The fuel shuffle is approximately 60% completed and currently underway. OPS 4 reports that Standby Service Water spray pond temperatures are both 79°F.

Which of the following is correct?

A. Enter TS LCO 3.0.3 immediately.

B. Restore control room AC subsystem to operable status in 30 days.

C. Suspend core alterations immediately.

D. No actions are required, LCO applicability not met.

ANSWER: C

QUESTION TYPE:	SRO
KA # & KA VALUE:	295018 2.1.12 – Partial or Complete loss of CCW. Ability to apply technical specifications for a system. IMP 4.0
REFERENCE:	TS 3.7.4 and bases and TS 3.7.1
SOURCE:	**NEW QUESTION –** T1, GP1
LO:	5226
RATING:	H2
ATTACHMENT:	YES – TS 3.7.4 and Bases; TS 3.7.1
JUSTIFICATION:	TS 3.7.4 bases states that SW and the UHS are part of the OPERABILITY requirements for CR HVAC to be operable. TS 3.7.1 requires the UHS temp to be LE 77°F. A is incorrect because TS 3.0.3 is not applicable. B is incorrect because both CR subsystems are inoperable. D is incorrect because TS 3.7.1 is applicable per TS 3.7.4 bases. C is correct.
COMMENTS:	Rewritten due to un-secure email sent.

Which of the following is **NOT** a basis for the low suppression pool water level Tech Spec?

Low suppression pool water level could result in…

A. less energy absorption and higher suppression pool temperatures following a DBA LOCA.

B. inadequate makeup water source for ECCS systems required following a DBA LOCA.

C. excessive clearing loads from SRV Tailpipe pipes during subsequent SRV actuations.

D. inadequate steam condensation from SRV quenchers during subsequent SRV actuations.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295030 2.4.21 – Knowledge of the parameters and logic used to assess the status of safety functions including: Containment conditions 55.43.2 IMP 4.3

REFERENCE: TS 3.6.2.2 Basis

SOURCE: **NEW QUESTION –** T1, GP1

LO: 6925

RATING: L3

ATTACHMENT: NONE

JUSTIFICATION: A, B, and D are all bases for the low suppression pool water level LCO and are incorrect. C is correct. It is a basis for the high suppression pool level.

COMMENTS:

Which of the following would **not** require a 50.59 screening to take place?

A. A plant modification made to the FPC Heat Exchanger service water outlet valve that allows it to be throttleable.

B. Installation of a jumper around the solenoid for ROA-V-1 which will prevent ROA-V-1 from closing on a loss of power.

C. Placing a portable heater in the SM-7 Switchgear Room during abnormally cold weather conditions to maintain operability.

D. Partial disassembly of DG-2 HVAC ducting to support repair of failed damper which is scheduled to be completed in one week.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 262001 2.2.8 – Knowledge of the process for determining if the proposed change /test/or experiment involves an unreviewed safety question 55.43.3 IMP 3.3

REFERENCE: 10CFR50.59 Resource Manual Pg 42; SWP-LIC-02 Rev. 4 page 3

SOURCE: **NEW QUESTION –** T2, GP1

LO: NONE

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: Disassembly of DG-2 ducting to support maintenance activities that are LT 90 days in duration do not require 50.59 review. The other three choices all require 50.59 review per references. D is correct.

COMMENTS: Rewritten per NRC comment.

The plant is operating at 32% power with control rod 30-31 selected. One of 4 operable A level LPRMs and all of the operable C level LPRMs feeding the RBM system fail downscale.

Which of the following is correct?

A. The failure of these LPRMs does not inop the RBM system.

B. All control rod movement must be suspended immediately except by scram.

C. RBM-A is not required to be operable until 35% power.

D. RBM-A has to be restored to operable status in 24 hours.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 215002 2.1.12 Ability to apply technical specifications for a system. 55.43.2 IMP 4.0

REFERENCE: SD000166 RBM rev. 10, page 5 and 8 TS 3.3.2.1

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5690, 7667

RATING: H3

ATTACHMENT: YES – TS 3.3.2.1

JUSTIFICATION: Center control rods provide inputs to each RBM from 8 LPRMs. Less than ½ the inputs causes a Rod Block and inops the RBM. A is incorrect. The RBM is required to be operable at 30% power so C is incorrect. The required action for one inop RBM at power is stated in D.

COMMENTS:

EX05014

The plant was operating at 99% power with OG-RIS-601A Post Treatment Rad Monitor, failed upscale. A failure on PP-8A-A results in a loss of power and a downscale on OG-RIS-601B Post Treatment Rad Monitor. All other plant equipment operates as expected.

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

A. PPM 5.2.1 Primary Containment Control will **not** be entered.

B. PPM 5.1.1 RPV Control will be entered for RPV level, pressure, and power control directions.

C. The plant continues to operate at a new lower power level due to increased backpressure.

D. Bypass flow would isolate and all OG flow is directed through the Offgas absorbers.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 271000A2.08 - Ability to predict the impacts of the following on the OFF GAS System; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: AC Dist failures 55.43.5 IMP 2.7

REFERENCE: LO000187 Off Gas rev. 10, pages 24-25

SOURCE: **NEW QUESTION –** T2, GP2

LO: 5621

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The loss describe causes an isolation of the Off Gas system. This results in increasing backpressure until the main turbine trips. C is incorrect. The reactor scrams and an entry will be made into PPM 5.1.1 on reactor level. B is correct. Vacuum continues to decrease until the MSIVs shut which causes suppression pool temp to increase and an entry into PPM 5.2.1. A is incorrect. With both rad monitors de-energized Offgas flow is isolated by OG-V-60 closing. D is incorrect. This system response is correct if just a hi radiation signal is received.

COMMENTS: Re-wrote distractor D to be credible.

EX05015

The plant was operating at 100% power with all plant equipment operating as required. A transient occurred which caused the following indication. No operator actions have been performed.

Reactor power	92% and stable
Reactor level	36 inches and stable
Core flow	100 mlbm/hr and stable
RRC Pump Speed	60 hz
RRC Loop A Flow	43,000 GPM
RRC Loop B Flow	44,000 GPM
JP Loop A Flow	43E6 lbm/hr
JP Loop B Flow	56E6 lbm/hr

Several feedwater heater level annunciators illuminated and cleared.

Which of the following actions is required?

A. Declare the loop with the lower flow to be “not in operation” in 2 hours.

B. Be in Mode 3 in 12 hours.

C. ABN-RRC-LOSS.

D. PPM 3.3.1 Reactor Scram.

ANSWER: B

QUESTION TYPE: SRO
KA # & KA VALUE: 290002 2.1.7 – Ability to evaluate plant performance and make operational judgments based on operation characteristics / reactor behavior / and instrument interpretation 55.43.2 IMP 4.4
REFERENCE: ABN-POWER rev. 4, page 2
SOURCE: **NEW QUESTION –** T2, GP2
LO: 6727
RATING: H3
ATTACHMENT: YES – TS 3.4.2
JUSTIFICATION: The indications given are from a failure of the #1 and #2 jet pumps. This is an entry condition for ABN-POWER and does not require an entry into any of the distracters. B is the only correct answer.
COMMENTS:

EX05016 FIXED TYPO ON ANSWER KEY

Maintenance is replacing the old refuel floor jib crane (1250 pounds) and wants to transport it over the spent fuel pool in order to remove it. The water level in the spent fuel pool is at the 605 feet elevation.

What are the limitations of this lift?

The load can be lifted to a maximum of …

A. 6 feet above the refuel floor.

B. 5 feet above the refuel floor.

C. 4 feet above the refuel floor.

D. 3 feet above the refuel floor.

ANSWER: ~~C~~ D

QUESTION TYPE:	SRO
KA # & KA VALUE:	2.2.26 Knowledge of refuel administrative requirements 55.43.7 IMP 3.7
REFERENCE:	LCS 1.9.2
SOURCE:	**BANK QUESTION –** T3
LO:	5362
RATING:	H2
ATTACHMENT:	LCS 1.9.2 and Figure 1.9.2-1
JUSTIFICATION:	The 1250 pound load can be lifted a maximum of 4 feet above the level of the spent fuel pool. Since the 606 elevation is 1 foot above the water level, the correct answer is 3 feet above the floor level. ~~C~~ D is correct.
COMMENTS:	

The plant is operating at 22% power in preparation for a refueling outage. A containment purge has just been initiated to de-inert the containment.

Which of the following is correct for this condition?

A. It is acceptable to use SGT with inoperable heaters for the purge because the heaters have no effect on the ability of the train used for the purge to perform its function.

B. The SGT train used for containment purge is inoperable due to the controller being placed in manual and no core alterations, operations with the potential for draining the reactor vessel, or movement of irradiated fuel is allowed.

C. The SGT train used for containment purge is inoperable due to the potential for rapid over pressurization prior to closure of the containment isolation valves following a LOCA.

D. At this power level, it is correct to use both trains of SGT for the containment purge because there is no postulated accident that can damage the Standby Gas Treatment System.

ANSWER: C

QUESTION TYPE: SRO
KA # & KA VALUE: 2.3.9 Knowledge of the process for performing a containment purge 55.43.4 IMP 3.4
REFERENCE: ODCM 6.2.2.6 and bases, TS 3.6.4.3
SOURCE: **NEW QUESTION –** T3
LO: 9498
RATING: H3
ATTACHMENT: YES - ODCM 6.2.2.6 and TS 3.6.4.3
JUSTIFICATION: The plant is in Mode 1, which requires that the purge be through 1 train of SGT for the first 24 hours. The train used for the purge, must be operable, which makes A incorrect. The direction is to use only 1 train at this power level, which makes D incorrect. With 1 train of SGT operable, there is a 7 day limit to return SGT to operable. Core alts, etc are allowed with only 1 train inop. B is incorrect. The basis for this RFO states that the train used for the purge is inoperable due to possible over pressurization during a LOCA. C is correct.
COMMENTS: Took ODCM bases out of attachment

The plant is operating at 99% power with a stuck open SRV. Suppression pool temperature is now 111° F and going up.

Which of the following is the appropriate procedure path used under these conditions?

Entry into…..

A. PPM 5.2.1 Primary Containment Control which requires entry into PPM 5.1.1 RPV Control which requires entry into PPM 3.3.1 Reactor Scram.

B. PPM 5.1.1 RPV Control and entry into PPM 5.2.1 Primary Containment Control which requires entry into PPM 5.1.3 Emergency RPV Depressurization.

C. PPM 5.1.1 RPV Control and entry into ABN-SRV and entry into PPM 5.2.1 Primary Containment Control which requires entry into PPM 5.1.3 Emergency RPV Depressurization.

D. PPM 5.2.1 Primary Containment Control which requires entry into PPM 5.1.1 RPV Control which requires entry into PPM 5.1.3 Emergency RPV Depressurization.

ANSWER: A

QUESTION TYPE: SRO

KA # & KA VALUE: 295026 2.4.16 – Knowledge of EOP implementation hierarchy and coordination with other support procedures 55.43.5 IMP 4.0

REFERENCE: PPM 5.1.1, 5.2.1, 5.1.3, 3.3.1, and ABN-SRV

SOURCE: **NEW QUESTION –** T1, GP1

LO: 8017

RATING: H3

ATTACHMENT: YES – 5.2.1 WW temp leg, PPM 5.1.1 Power leg. HCTL curves.

JUSTIFICATION: The conditions given require EOP entries in PPM 5.2.1 Primary Containment Control, which directs that PPM 5.1.1 RPV Control be entered before 110° F. 5.1.1 directs exiting to 3.3.1 with all control rods in. There are no conditions given which require an ED and ABN-SRV actions are superceded by the directions given in the EOPs . A is correct.

COMMENTS: Rewritten per NRC comments.

EX05019

The plant was operating at 75% power when a transient occurred that resulted in a loss of feedwater. Control rod 30-31 did not fully insert. None of HPCS, CRD, and RCIC could be started and level has been returned by low pressure systems to the normal operating band following Emergency Depressurization as directed by the EOPs. All other plant systems operated as expected.

Which of the following is correct concerning these conditions?

As the Emergency Director, declare a(n)…

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. General Emergency.

ANSWER: C

QUESTION TYPE: SRO

KA # & KA VALUE: 295031 2.4.38 – Ability to take actions called for in the facility emergency plan / including supporting or acting as emergency director 55.43.5 IMP 4.0

REFERENCE: PPM 13.1.1

SOURCE: **NEW QUESTION –** T1, GP1

LO: 6131

RATING: H3

ATTACHMENT: YES – PPM 13.1.1 pages 14 and 15

JUSTIFICATION: The loss of high pressure systems results in reactor level being reduced to TAF and ED to allow LP systems to inject. One control rod not inserting by itself does not require entry into the ATWS procedure. Therefore, the correct answer is C SAE was declared due to 2.1.S.1.

COMMENTS: Revised stem to specifically say we ED'ed to eliminate having reduced RPV pressure to feed with Condensate pumps which may have prevented RPV/L from dropping below TAF and getting a different answer.

A control rod withdrawal is in progress for a plant start up. Power is currently 250,000 counts on SRM A and C. SRM B and D indicate 95,000 counts.

Which of the following is correct for this condition?

A. Enter PPM 4.603.A7.3-4 SCRAM SYSTEM A.

B. Enter PPM 4.603.A7.2-7 ROD OUT BLOCK.

C. Enter PPM 3.3.1 Reactor Scram.

D. Control rod withdrawal can continue.

ANSWER: B

QUESTION TYPE: SRO

KA # & KA VALUE: 215004A2.04 – Ability to predict the impacts of the following on the SRM System; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale and downscale trips 55.43.5 IMP 3.7

REFERENCE: SD000132 rev 10, page 26

SOURCE: **NEW QUESTON –** T2, GP1

LO: 5795

RATING: H2

ATTACHMENT: NONE

JUSTIFICATION: The count level given for both A and C is above the scram setpoint on 2E5 counts. Since the shorting links are installed for a normal startup, the scram is not in affect. A and C are incorrect. The indications are above the rod block setpoint of 1.0E5 counts so D is incorrect. Since the rod block setpoint has been exceeded, B is correct.

COMMENTS:

EX05021

The plant is operating at 21% power following an outage. During the outage maintenance was performed on 6 SRVs with setpoints of 1165 psig and 1175 psig. These SRVs have not yet been declared operable.

Which of the following is correct concerning these conditions?

A. Action must be taken within 1 hour to place the plant in Mode 2 within 7 hours and Mode 3 within 13 hours.

B. The Mode Switch must be placed in SHUTDOWN immediately.

C. Power can be increased to GT 25% as long as 2 of the inoperable SRVs are declared operable within 4 hours of exceeding 25% power.

D. Power can be increased to 24% without declaring the SRVS operable.

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 239002 2.2.21 – Knowledge of pre and post maintenance operability requirements 55.43.2 IMP 3.5

REFERENCE: TS 3.4.3, 3.4.4, 3.0.4

SOURCE: **NEW QUESTON -** T2, GP1

LO: 10306

RATING: H3

ATTACHMENT: YES - TS 3.4.3, 3.4.4, 3.0.4

JUSTIFICATION: We have 18 SRVs and TS 3.4.3 requires only 12 to be operable for the safety function at GT 25% power. However, 3.4.3 requires that at least 2 be operable in the lowest 2 lift groups. The question states that there are no operable SRVs in the lowest 2 lift groups. TS 3.0.4 states that the specified condition of the operability must be allowed for an indefinite time for 3.0.4 to be utilized to go above 25% power (the condition is limited for 4 hours in this case). There is no 3.0.3 condition given in the question. Therefore, only D is correct. .

COMMENTS: Revised justification from 3.4.4 to 3.4.3 in second sentence.

EX05022

According to the Columbia Generating Station Facility Operating License, what is the maximum licensed power level?

A. 3323 mwt

B. 3386 mwt

C. 3423 mwt

D. 3486 mwt

ANSWER: D

QUESTION TYPE: SRO

KA # & KA VALUE: 2.1.10 Knowledge of conditions and limitations in the facility license. 55.43.1 IMP 3.9

REFERENCE: Columbia License

SOURCE: **NEW QEUSTION –** T3

LO: 10296

RATING: L2

ATTACHMENT: NONE

JUSTIFICATION: According to the facility operating license, 3486 is the max license power level. A is the old max level and B/C are incorrect combinations of the correct numbers.

COMMENTS: This question was added after deletion of the old 05022 to more evenly balance the exam to the requirements of 10CFR 55.43. The outline will be updated to reflect this change.

EX00100

The plant is operating at 97% power with a discharge from the Waste Collector Tank to the Circ. Water Blowdown line underway. An annunciator is received in the control room indicating that Process Rad Monitor FDR-RIS-606 (Radwaste Effluent) has failed 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} micro curie/ml, at least once every 12 hours.

B. The discharge may continue for up to 30 days provided that the discharge flow rate 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 Knowledge of SRO responsibilities for auxiliary system that are outside the control room (e.g. / waste disposal and handling systems) 55.43.4

IMP 2.9

REFERENCE: ODCM 6.1.1 table 6.1.1.1-1

SOURCE: **BANK QUESTION from 2000 NRC exam – Slightly modified for clarification -** T3

LO: 7721 5650

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 flow rate 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:

The plant is shutdown in Mode 5 with a complete core offload in progress. The outside air temperature is 110° F. Ops 2 reports that the general area temperature in the 471 RB west end (around E-SH-10) has been 105° F for the last 75 minutes. The temperature is due to maintenance on the ventilation system in the local area. All other plant equipment is operating normally.

Which of the following actions is correct for this condition?

A. Immediately suspend movement of irradiated fuel in the secondary containment, core alterations, and initiate actions to suspend operations with a potential for draining the reactor vessel.

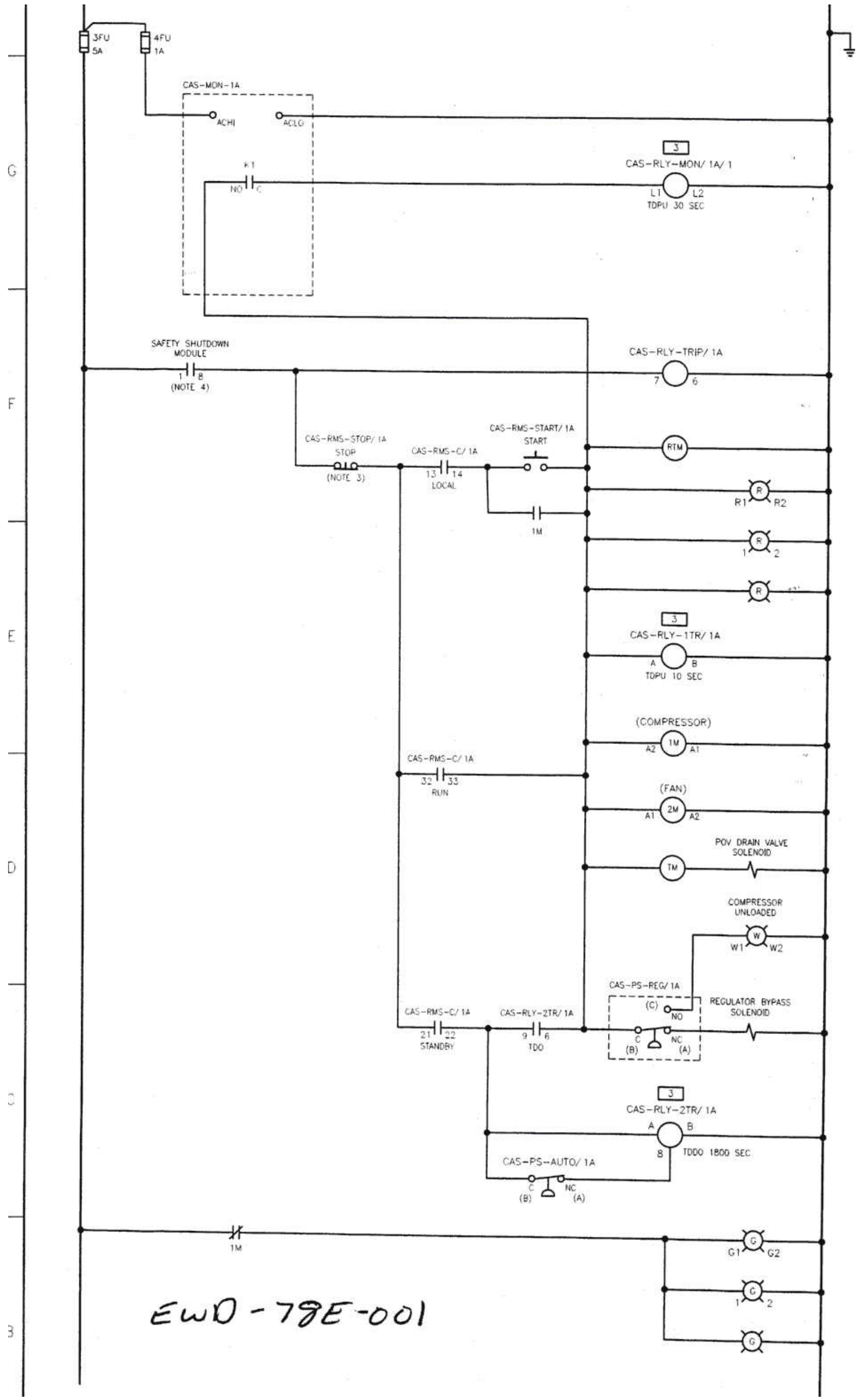
B. The actions required can be delayed for a maximum of 4 hours due to the temperature increase resulting from a maintenance issue.

C. Initiate action to restore the area to within the limits of Condition B in 4 hours.

D. Initiate action to restore the area to within the limits of Condition C in 4 hours.

ANSWER: A

QUESTION TYPE:	SRO
KA # & KA VALUE:	295032EA2.02 - Ability to determine and/or interpret the following a they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Equipment Operability 55.43.2 IMP 3.5
REFERENCE:	LCS 1.7.1 and TS 3.6.4.3
SOURCE:	**NEW QUESTION –** T1, GP2
LO:	9540
RATING:	H4
ATTACHMENT:	YES – LCS 1.7.1 pages -1-3, 8, 15, &16 and TS 3.6.4.3 pages -1 - 3
JUSTIFICATION:	This high temperature for 75 minutes causes SGT div 1 and 2 to be inop under these conditions. The extra allotted 4 hours does not apply when Condition C is exceeded or the Condition B is exceeded for maintenance activities. B is incorrect. C and D are both incorrect because the allotted time would be 1 hour. A is correct per TS 3.6.4.3
COMMENTS:	


3FU
5A
4FU
1A
CAS-MON-1A
ACHI
ACLO
K1
NO
C
3
CAS-RLY-MON/ 1A/ 1
L1
L2
TDPU 30 SEC
SAFETY SHUTDOWN
MODULE
1
8
(NOTE 4)
CAS-RLY-TRIP/ 1A
7
6
CAS-RMS-STOP/ 1A
STOP
(NOTE 3)
CAS-RMS-C/ 1A
13
14
LOCAL
CAS-RMS-START/ 1A
START
1M
RTM
R
R1
R2
1
2
3
CAS-RLY-1TR/ 1A
A
B
TDPU 10 SEC
(COMPRESSOR)
1M
A2
A1
CAS-RMS-C/ 1A
32
33
RUN
(FAN)
2M
A1
A2
POV DRAIN VALVE
SOLENOID
TM
COMPRESSOR
UNLOADED
W
W1
W2
CAS-PS-REG/ 1A
(C)
NO
C
NC
(B)
(A)
REGULATOR BYPASS
SOLENOID
CAS-RMS-C/ 1A
21
22
STANDBY
CAS-RLY-2TR/ 1A
9
6
TDO
3
CAS-RLY-2TR/ 1A
A
B
8
TDDO 1800 SEC
CAS-PS-AUTO/ 1A
C
NC
(B)
(A)
1M
G
G1
G2
1
2
G
F
E
D
C
B

EWD-78E-001

RO Handout

1 To prevent taking action based on erroneous RPV level indication, an RPV level instrument may not be used to determine RPV level if any of the following conditions exist for that instrument.

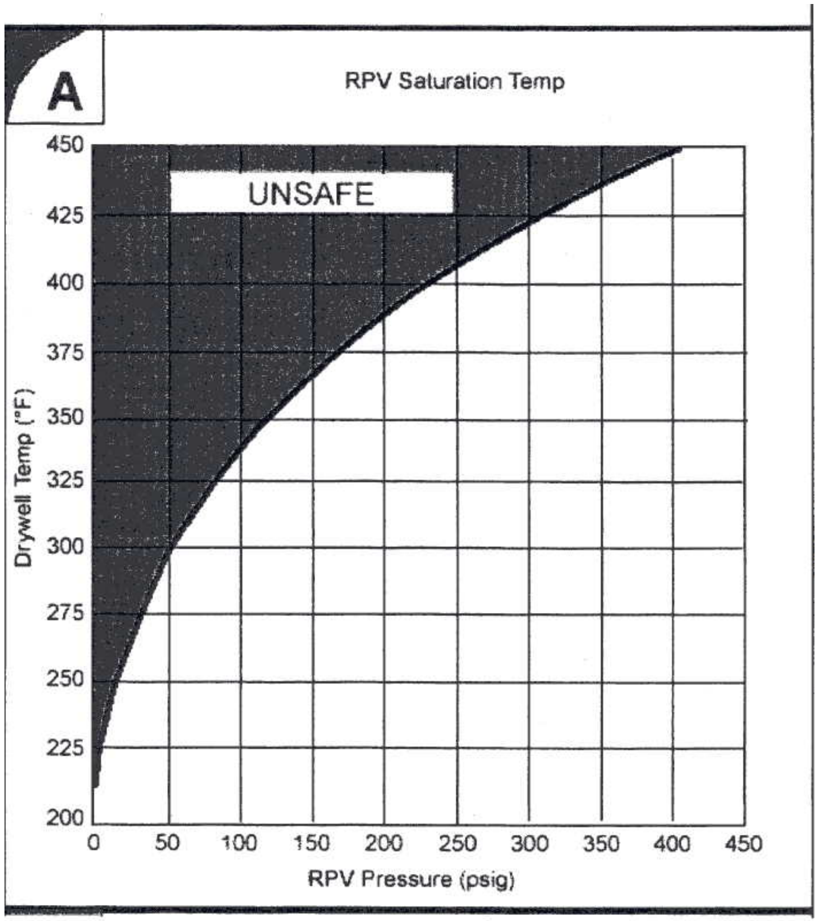
a. Drywell temp is at or above RPV Saturation Temp. and erroneous / erratic indication is observed. A

b. The instrument is identified as unusable per ABN-HELB or ANB-INSTRUMENTATION

c. For any of the instruments in the following table, the instrument reads below the Minimum Usable Level.

Instrument	Range (in.)	Drywell Temp Range (°F)	Minimum Usable Level (in.)
Wide Range	-150 to +60	100 - 550	-147
Fuel Zone Range	-310 to -110	100 - 550	-285
Narrow Range	0 to +60	100 - 550	+ 4
Upset Range	0 to +180	100 - 160 161 - 200 201 - 250 251 - 300 301 - 350 351 - 400 above 400	+0 +13 +33 +57 +84 +114 +180
Shutdown Flooding Range	0 to +400	100 - 150 151 - 200 201 - 250 251 - 300 301 - 350 351 - 400 above 400	+19 +32 +47 +65 +86 +109 +206

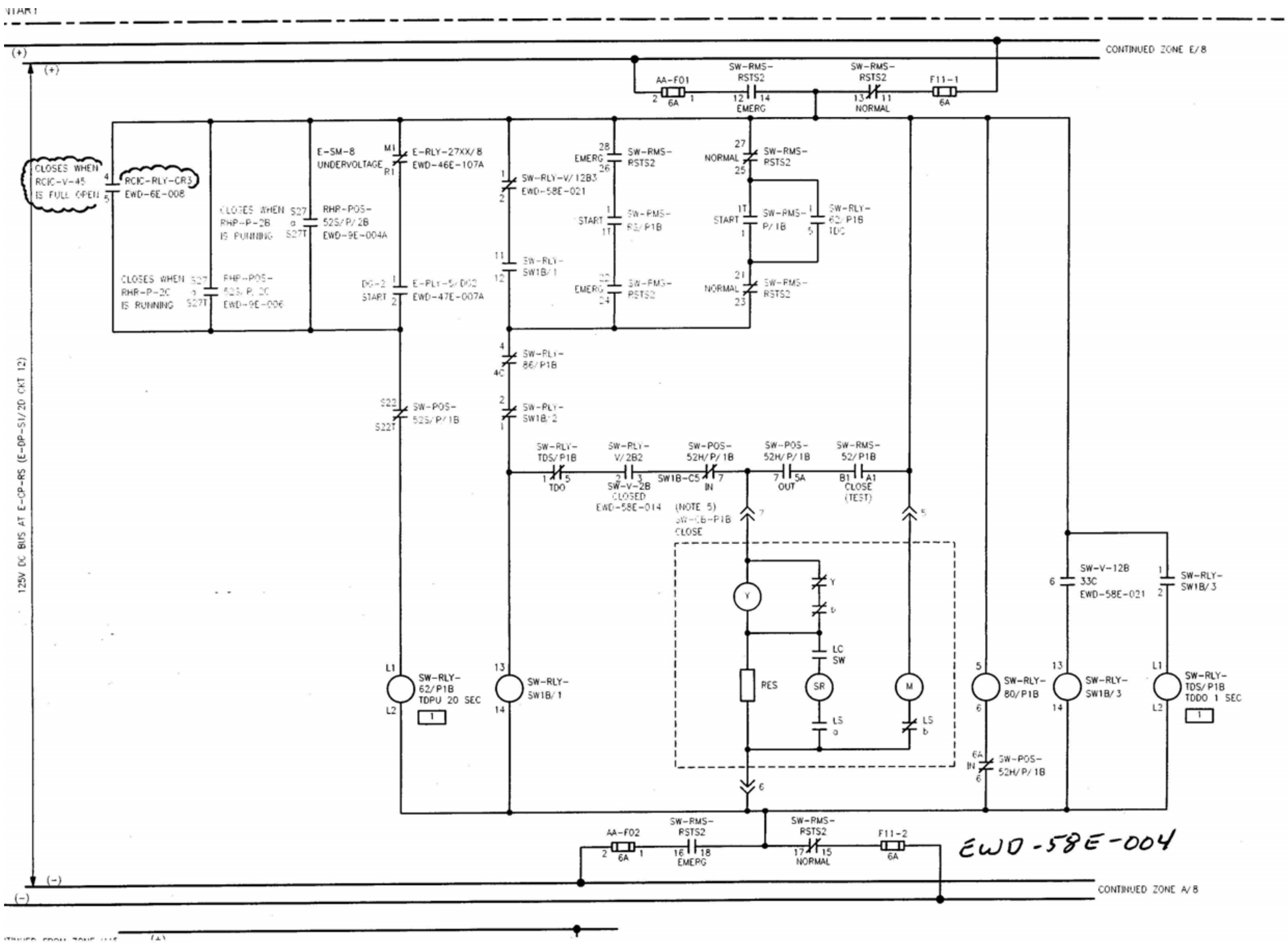
PPM 5.1.1


A
RPV Saturation Temp
UNSAFE
Drywell Temp (°F)
450
425
400
375
350
325
300
275
250
225
200
0
50
100
150
200
250
300
350
400
450
RPV Pressure (psig)

PPM 5.1.1

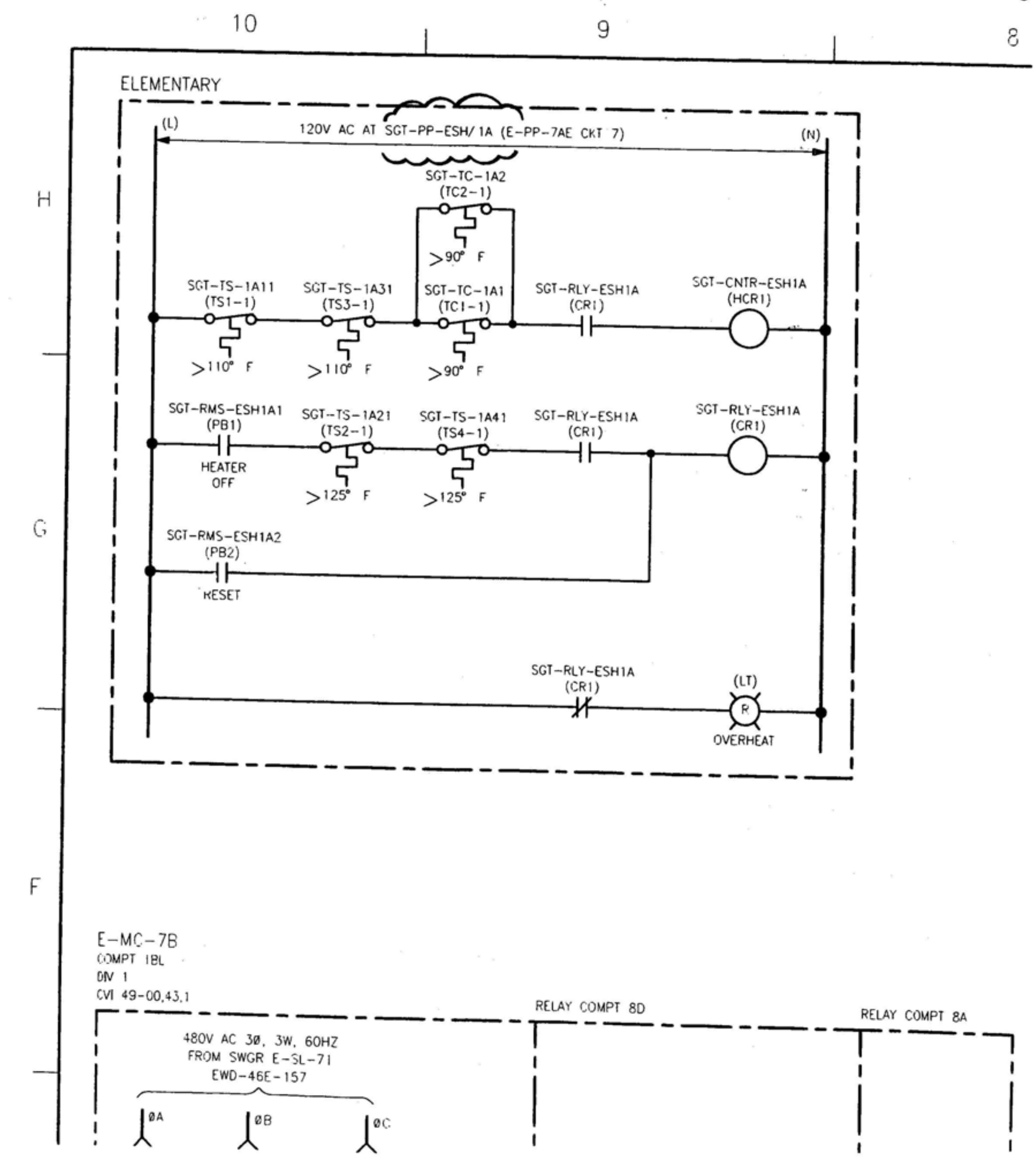
25	RB Area Water Levels	
Area	Alarm Level (in. above floor)	Maximum Safe Operating Value PPM 5.5.27 (in. above floor)
LPCS Pump Room	6	58
HPCS Pump Room	6	69
RHR Pump Room A	6	36
RHR Pump Room B	6	72
RHR Pump Room C	6	67
RCIC Pump Room	6	6

PPM 5.3.1

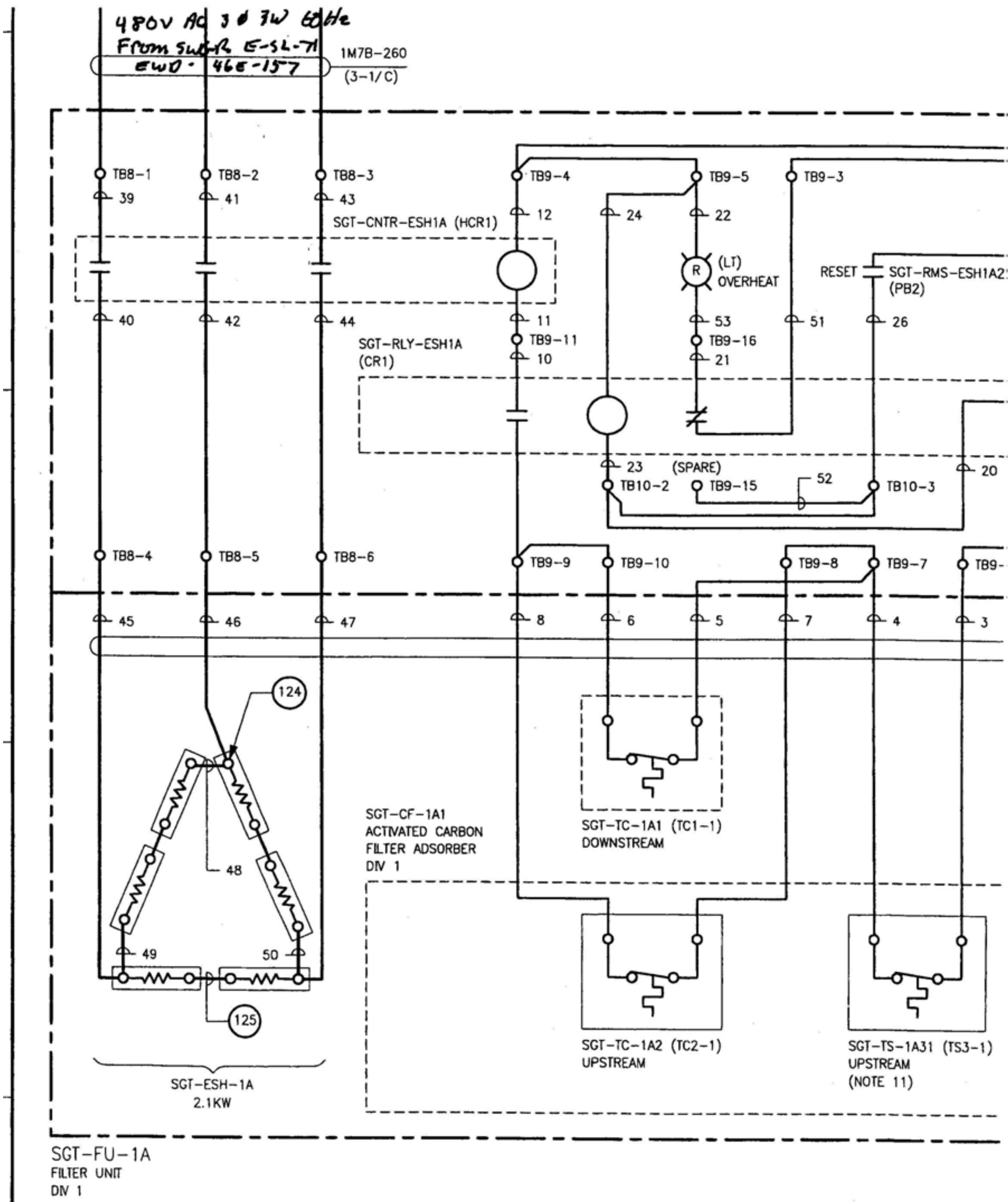


125V DC BUS AT E-CP-RS (E-DP-S1/2D CKT 12)
CONTINUED ZONE E/8
CONTINUED ZONE A/8
AA-F01
6A
SW-RMS-RSTS2
EMERG
NORMAL
F11-1
CLOSES WHEN RCIC-V-45 IS FULL OPEN
RCIC-RLY-CR3
EWD-6E-008
CLOSES WHEN RHP-P-2B IS RUNNING
RHP-POS-52S/P/2B
EWD-9E-004A
CLOSES WHEN RHR-P-2C IS RUNNING
EWD-9E-006
E-SM-8 UNDERVOLTAGE
E-RLY-27XX/8
EWD-46E-107A
DG-2 START
E-RLY-5/DG2
EWD-47E-007A
SW-POS-52S/P/1B
SW-RLY-V/12B3
EWD-58E-021
SW-RLY-SW1B/1
SW-RLY-86/P1B
SW-RLY-SW1B/2
SW-RMS-52/P1B
START
SW-RMS-P/1B
SW-RLY-62/P1B
TDC
SW-RLY-TDS/P1B
TDO
SW-RLY-V/2B2
SW-V-2B CLOSED
EWD-58E-014
SW1B-CS
SW-POS-52H/P/1B
IN
OUT
CLOSE (TEST)
(NOTE 5)
SW-CB-P1B CLOSE
RES
SR
LC SW
LS
M
SW-RLY-62/P1B TDPU 20 SEC
SW-RLY-SW1B/1
SW-RLY-80/P1B
SW-POS-52H/P/1B
SW-V-12B 33C
EWD-58E-021
SW-RLY-SW1B/3
SW-RLY-TDS/P1B TDDO 1 SEC
AA-F02
F11-2
EWD-58E-004

EWD-39E-004



10
9
8
H
G
F
ELEMENTARY
(L)
120V AC AT SGT-PP-ESH/1A (E-PP-7AE CKT 7)
(N)
SGT-TC-1A2
(TC2-1)
>90° F
SGT-TS-1A11
(TS1-1)
>110° F
SGT-TS-1A31
(TS3-1)
>110° F
SGT-TC-1A1
(TC1-1)
>90° F
SGT-RLY-ESH1A
(CR1)
SGT-CNTR-ESH1A
(HCR1)
SGT-RMS-ESH1A1
(PB1)
HEATER
OFF
SGT-TS-1A21
(TS2-1)
>125° F
SGT-TS-1A41
(TS4-1)
>125° F
SGT-RLY-ESH1A
(CR1)
SGT-RLY-ESH1A
(CR1)
SGT-RMS-ESH1A2
(PB2)
RESET
SGT-RLY-ESH1A
(CR1)
(LT)
R
OVERHEAT
E-MC-7B
COMPT 1BL
DIV 1
CVI 49-00,43,1
RELAY COMPT 8D
RELAY COMPT 8A
480V AC 3Ø, 3W, 60HZ
FROM SWGR E-SL-71
EWD-46E-157
ØA
ØB
ØC



REV	DATE	DESCRIPTION	DWN	CHK

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	$\leq$ 0.58W + 51% RTP
b. Inop	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	NA
c. Downscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5	$\geq$ 3% RTP
2. Rod Worth Minimizer	$1^{(b)}, 2^{(b)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA
3. Reactor Mode Switch – Shutdown Position	(c)	2	SR 3.3.2.1.7	NA

(a) THERMAL POWER $\geq$ 30% RTP and no peripheral control rod selected.

(b) With THERMAL POWER $\leq$ 10% RTP.

(c) Reactor mode switch in the shutdown position.

<u>**SUMMARY OF REPORTING CRITERIA**</u>

Regulatory Reference	Requirement/Time Limit/Recipient
10 CFR 50.72(b)(2)(iv)(B)[1]	Reactor Protection System Actuation when the reactor is critical except when it is preplanned during testing or reactor operation. [1]4 hr. - NRC *
10 CFR 50.72(b)(3)(xii)[1] 10 CFR 72.75(c)(3)[1] 10 CFR 40.60(b)(3)[2,4]	Transport of a Radiologically Contaminated Person to an offsite medical facility for treatment. [1]8 hr. - NRC [2]24 hrs. - NRC [4]30 days - NRC
10 CFR 50.72(b)(2)(xi)[1] 10 CFR 72.75(b)(2)[1]	News Release or Notification to Other Government Agency, either planned or made, concerning an event related to the health and safety of the public or on-site personnel or protection of the environment. [1]4 hr. - NRC*
10 CFR 72.75(c)(1)[1,2] 10 CFR 72.75(c)(2)[1,2]	Defect in or reduction in effectiveness of Spent Fuel Storage Cask. [1]8 hr. - NRC [2]60 days - NRC
10 CFR 50.72(b)(3)(iv)(A)[1]	Valid actuation of any of the systems listed in 10CFR50.72(b)(3)(iv)(B) except when the actuation is preplanned during testing or reactor operation. See NUREG 1022, Revision 2, Section 3.2.6. [1]8 hr. - NRC
10 CFR 40.60(a)[1, 2]	Condition that prevents actions to avoid exposure to radiation/radioactive material. [1]4 hrs. - NRC [2]30 days - NRC *

* CEO notification required per GIH-9.1.3.

Attachment 7.2

1.10.1

SRO Handout

ENERGY NORTHWEST
People · Vision · Solutions

COLUMBIA GENERATING STATION
PLANT PROCEDURES MANUAL

USE CURRENT REVISION

NUMBER	APPROVED BY	DATE
*1.10.1	DWC - Revision 26	10/04/04

VOLUME NAME

ADMINISTRATIVE PROCEDURES

SECTION

PLANT REPORTING REQUIREMENTS

TITLE

NOTIFICATIONS AND REPORTABLE EVENTS

SUMMARY OF REPORTING CRITERIA

Regulatory Reference	Requirement/Time Limit/Recipient
10 CFR 72.75(c)(2)[1, 2] 10 CFR 40.60(b)(2)[1, 3]	Event in which equipment important to safety is disabled or fails to function as designed. [1]24 hrs. - NRC (or by 8:00 am eastern time on next business day) [2]60 days - NRC [3]30 days - NRC
10 CFR 50.73(a)(iii)[1]	Actual Threat (Natural or External) to Plant or Personnel Safety [1]60 days - NRC
10 CFR 70.52[1] 10 CFR 72.74(a)[1,2]	Accidental Criticality or any loss of special nuclear material [1]1 hr. - NRC [2] 30 days - NRC *
10 CFR 50.72(b)(3)(v)[1] 10 CFR 50.73(a)(2)(v)[2]	Condition that could have prevented Safe Shutdown, Residual Heat Removal, Control of Radiation Release, or Accident Mitigation [1]8 hrs. - NRC [2]60 days - NRC
ODCM 6.2.2.5	Ventilation Exhaust Treatment System Inoperable for More than 31 Days 10 days - NRC
Tech. Spec. 3.3.3.1 Tech. Spec. 5.6.6	Primary Containment Area Radiation Monitor Channels Inoperable for More than 30 Days (One Channel Inoperable) or 7 Days (Two Channels Inoperable) 14 days - NRC
10 CFR 40.60(b)(4)[1, 2]	Unplanned fire or explosion damaging any licensed material or any device, container, or equipment containing licensed material when quantity is five times greater than limits or damage affects integrity. [1]24 hrs. - NRC [2]30 days - NRC

* CEO notification required per GIH-9.1.3.

Attachment 7.2

3-3 ENGINE 1 LUBE OIL LEVEL LOW

3-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
ENG. 1 LUBE OIL LEVEL LOW	DLO-LS-1B1 LE 400 gallons (K35)	None

> CAUTION: Operation of the engine with oil sump level less than the LOW mark (LT 230 gallons) on the dipstick may result in severe damage to the engine.

NOTE: Oil sump level is required to be GT 9 inches above the LOW mark on the dipstick to meet the operability requirements of Technical Specification 3.8.3.

1. Check engine lube oil sump level.
2. Verify oil level is GT 9 inches above the LOW mark on the dipstick.
3. Notify the CRS/Shift Manager of the oil level.
4. Check engine lube oil sump level every hour until the alarm clears or the engine is shut down.
5. Have Maintenance fill the engine lube oil sump to the appropriate FULL mark (shutdown or running) on the dipstick.
6. Refer to Technical Specification 3.8.3 for required actions.
7. Contact system engineer.

REFERENCES: E543-2VB-2
M512, Sh 3

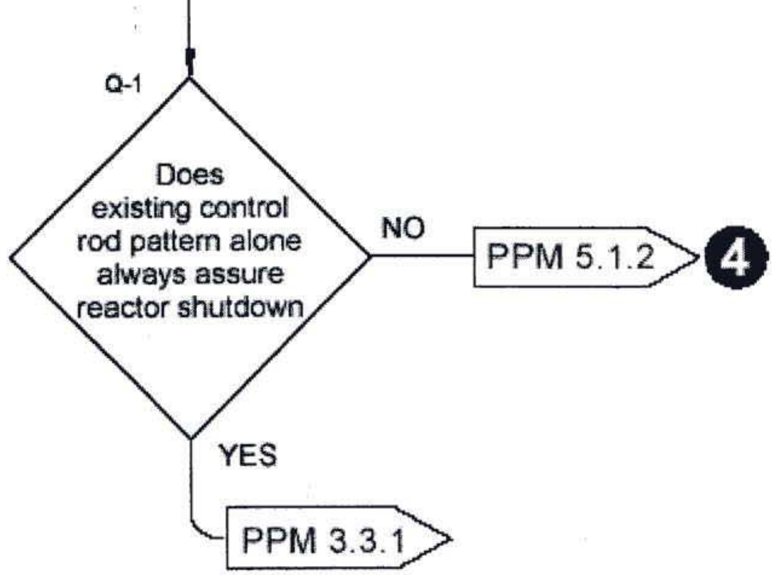
6-6 RADWASTE EFFLUENT RADIATION MONITORS DOWNSCALE OR INOPERABLE

6-6 WINDOW	SOURCE	AUTOMATIC ACTIONS
RADWASTE EFFLUENT RAD MONITORS DNSCL OR INOP	FDR-RIS-606 (LE 0.5 cps)	None

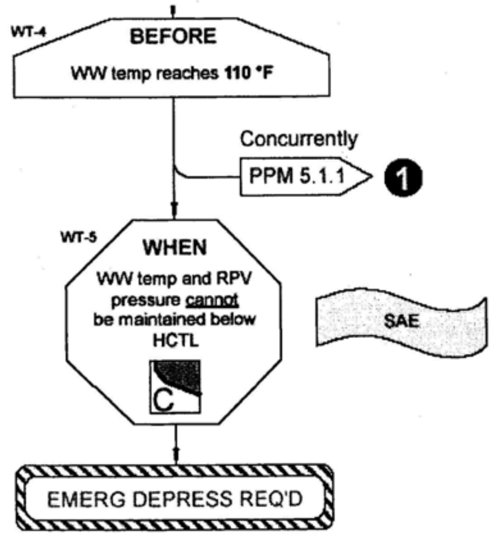
1. Confirm downscale condition of FDR-RIS-606 and/or SW-RR-1 Pen 1 (Red) (H13-P604).
2. Check mode switch in OPERATE position.
3. Secure radwaste discharge to river, if in progress, until monitor returned to operable status.
4. Refer to ODCM 6.1.1.1 if a discharge to the river was in progress.
5. Attempt to reset FDR-RIS-606.
6. If FDR-RIS-606 is de-energized, check power supply at:
   a. E-DP-S0/A, ckt 2, or
   b. H13-P604, BB-F1 and BB-F2.

REFERENCES: CVI: 02D17-05,34, Sh 3, 7
EWD-36E-0001, 0007

Reactor Power

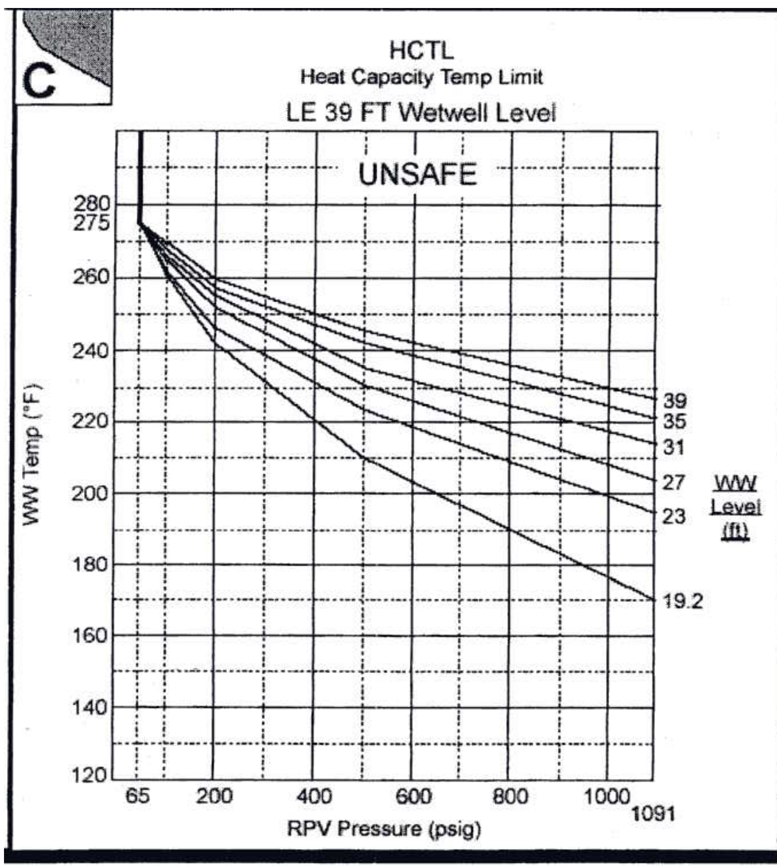


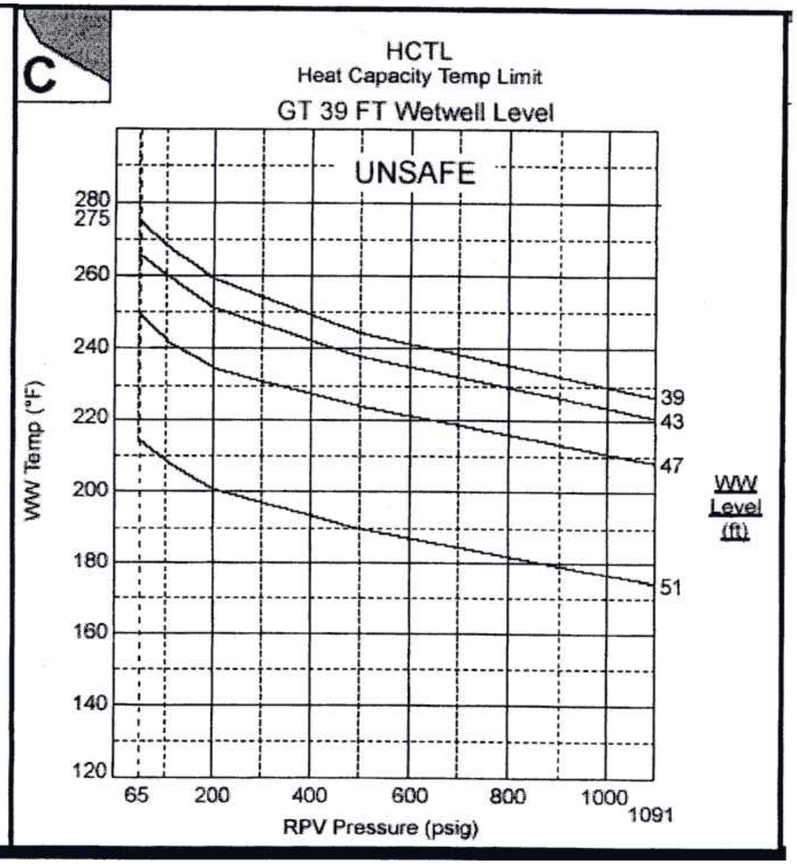
PPM 5.1.1



WT-4
BEFORE
WW temp reaches 110 °F
Concurrently
PPM 5.1.1
1
WT-5
WHEN
WW temp and RPV pressure cannot be maintained below HCTL
C
SAE
EMERG DEPRESS REQ'D

PPM 5.2.1


C
HCTL
Heat Capacity Temp Limit
LE 39 FT Wetwell Level
UNSAFE
WW Temp (°F)
280
275
260
240
220
200
180
160
140
120
65
200
400
600
800
1000
1091
RPV Pressure (psig)
39
35
31
27
23
19.2
WW Level (ft)


C
HCTL
Heat Capacity Temp Limit
GT 39 FT Wetwell Level
UNSAFE
WW Temp (°F)
280
275
260
240
220
200
180
160
140
120
65
200
400
600
800
1000
1091
RPV Pressure (psig)
39
43
47
51
WW Level (ft)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2 RPV 2.1 RPV Water Level	RCS Leakage **2.1.U.1** 1 2 3 Valid unidentified leakage GE 10 gpm or upscale high indicated on recorder EDR-FRS-623, Pen 1 (P632) (Non RCC) OR Valid identified leakage GE 25 gpm indicated on recorder EDR-FRS-623, Pen 2 (P632)	Loss OR potential loss of RCS **2.1.A.1** 1 2 3 Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high	Loss or potential loss of any two fission product barriers **2.1.S.1** 1 2 3 4 5 RPV level LT -161 inches (for ATWS conditions, RPV level LT -183") or cannot be determined	A loss of any two fission product barriers and loss of potential loss of the third **2.1.G.1** 1 2 3 Entry into Severe Accident Guidelines **2.1.G.2** 1 2 3 RPV level LT -161 inches, (for ATWS conditions, RPV level LT -183") or cannot be determined AND ANY of following: • Rapid unexplained decrease of PC pressure following an initial increase. • Drywell pressure response not consistent with LOCA conditions • Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists

Attachment 5.1

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
2 RPV 2.2 Reactivity Control	Inadvertent Criticality **2.2.U.1** 2 3 4 5 An extended and unplanned sustained positive period observed on NIs, while NOT performing a reactor startup.	Failure of Reactor Protection System (RPS) instrumentation to complete or initiate a reactor scram AND manual scram was successful. **2.2.A.1** 1 2 Any RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND Manual Actions (mode switch in shutdown, manual push buttons and ARI) result in reactor power LE 5%.	Failure of RPS instrumentation to complete or initiate an automatic reactor scram once a RPS setpoint has been exceeded AND manual scram was NOT successful **2.2.S.1** 1 2 Any RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND either: Reactor power GT 5% or unknown OR Wetwell temperature GT 110°F	Failure of the RPS to complete an automatic scram AND manual scram was NOT successful AND there is indication of an extreme challenge to the ability to cool the core. **2.2.G.1** 1 2 Any RPS setpoint (including manual) has been exceeded per Technical Specification 3.3.1.1 AND RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions AND Wetwell temperature cannot be maintained LT the HCTL

Attachment 5.1

COLUMBIA GENERATING STATION EMERGENCY CLASSIFICATION TABLE

TABLES 4 & 5

cps = counts per second   cpm = counts per minute   PMU = panel meter units   N/A = not applicable (outside of meter range)

Table 4 Offsite Dose Calculation/Field Survey Sample Analysis Classification Thresholds at 1.2 miles

	UE	Alert	Site Area	General
TEDE	N/A	N/A	100 mrem	1000 mrem
CDE Thyroid	N/A	N/A	500 mrem	5000 mrem
TEDE rate	0.1 mrem/hr	10 mrem/hr	100 mrem/hr (projected GT 60 min)	1000 mrem/hr (projected GT 60 min)
CDE Thyroid rate	0.3 mrem/hr	50 mrem/hr	500 mrem/hr (for GT 1 hr inhalation)	5000 mrem/hr (for GT 1 hr inhalation)

Table 5 Safe Shutdown Buildings
• Vital portions of the RadWaste/Control Building • Reactor Building • Vital portions of the Turbine Building • Standby Service Water Pump Houses • Diesel Generator Building • Diesel Generator Fuel Oil Storage Area

TABLE 6 FISSION PRODUCT BARRIER DEGRADATION TABLE

Fuel Clad Loss	Fuel Clad Potential Loss	RCS Loss	RCS Potential Loss	PC Loss	PC Potential Loss
Coolant activity GT 300 μCi/gm dose equivalent iodine	RPV level LT -161 inches (for ATWS conditions, RPV level LT -183 inches)	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 70 R/hr	Total RCS leakage GT 30 gpm inside PC or EDR-FRS-623, Pen 2 upscale high	Rapid unexplained decrease of PC pressure following an initial increase	Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 14,000 R/hr
Containment Radiation Monitor CMS-RIS-27E and CMS-RIS-27F reading GT 3,600 R/hr		RPV level LT -161 inches (for ATWS conditions, RPV level LT -183 inches)	Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1, "Secondary Containment Control")	Drywell pressure response not consistent with LOCA conditions	PC H_2 and O_2 concentrations GT 6% H_2 and 5% O_2
Entry into Severe Accident Guidelines		Drywell pressure GT 1.68 psig with indications of RCS leakage inside drywell		Failure of containment isolation valves (LCS Table 1.6.1.3-1) in any one line to close following auto or manual initiation AND downstream pathway outside primary containment exists OR Unisolable primary system discharging outside PC resulting in any area temperature or radiation level above Maximum Safe Operating Values (PPM 5.3.1,"Secondary Containment Control")	Entry into Severe Accident Guidelines Loss of pressure suppression function Cannot maintain plant parameters within HCTL or SRVTPLL
				Intentional venting per PPM 5.2.1, "Primary Containment Control"	Wetwell pressure exceeds PSP PC pressure exceeds PCPL
Any event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of the fuel clad barrier		Any event, in the judgement of the Emergency Director, that could lead or has led to a loss or potential loss of the RCS barrier		Any event, in the judgement of the Emergency Director, that could lead to or has led to a loss or potential loss of primary containment barrier	

ENERGY NORTHWEST
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GENERAL INFORMATION HANDBOOK
ADMINISTRATIVE PROCEDURES MANUAL

NUMBER	APPROVED BY	DATE
GIH-9.1.3	J. V. Parrish - Revision 0	06/14/04

CHAPTER
COMMUNICATION

TITLE
CHIEF EXECUTIVE OFFICER EVENT NOTIFICATION

1.0 PURPOSE AND SCOPE

This procedure provides Energy Northwest personnel with guidance on what events require the Chief Executive Officer (CEO) be notified and the maximum time from the initiation of the event until the CEO has been notified of its occurrence. The events are detailed in the attachment to this procedure.

2.0 RESPONSIBILITIES

2.1 Senior Staff

The cognizant senior staff member or his/her designee will use the attachment to determine when the CEO is to be notified for the specific events listed. There may be other instances, not currently identified in the attachment, in which the senior staff member may wish to notify the CEO of an event. In this case, the time from the occurrence of the event to notification is at the discretion of the responsible senior staff member. The senior staff member should ask themselves "If I were the CEO would I want to know this?" If the answer is "yes," then call the CEO, if the answer is "no," then do not call. If the answer is "I'm not sure," then call the CEO.

2.2 Managers/Supervisors

Managers and supervisors cognizant of the event, as delineated in the attachment, will notify the senior staff member in a timely manner such that the senior staff member can meet the specified time requirements.

2.3 Employees/Contractor Personnel

Employees or contractor personnel involved in an event as delineated in the attachment, will notify the manager or supervisor in a timely manner such that the manager can notify the senior staff member to allow them to meet the specified time requirements.

3.0 ATTACHMENTS

3.1 Matrix of Events and Notification Requirements

MATRIX OF EVENTS AND NOTIFICATION REQUIREMENTS

Category	Event	Time to Notify
HR	Employee death at work	1 Hour
Operational	An event that requires an emergency classification of Unusual Event or higher	1 Hour
Operational	Any unplanned entry into a shutdown Technical Specification of 7 days or less.	1 Hour
Safety	Natural disaster that affects Energy Northwest facilities	1 Hour
PR	Workplace accidents or incidents that may result in adverse publicity	1 Hour
Security	Bomb threat	1 Hour
Security	Security violation – One Hour Report	1 Hour
Safety	"Man down" or emergency response (ambulance, fire) event	1 Hour
Operational (10CFR72)	Any event that results in a 1 or 4 hour reportable condition.	2 Hours
Operational	NSIP complaint that may impact Nuclear Safety	4 Hours
Operational	An issue raised verbally by the NRC, INPO or other regulatory agency if judged to be significant	4 hours
Operational	Any unplanned reduction in power at the Columbia Generating Station	4 Hours
Operational	Significant human performance event	4 Hours
Operational	Significant operational event/condition/issue	4 Hours
Operational	Any unplanned entry into Technical Specifications that is discovered after the fact. (8 Hour Reportable)	4 Hours
Operational	Any station licensed or non-licensed operator removed from watch standing or disqualified.	4 Hours
PR	Inquiries from the media of a sensitive nature	4 Hours
Safety	An event that can be categorized as a significant near miss	4 Hours
Safety	Industrial injury or accident that could lead to an OSHA recordable event, restricted duty or lost time	4 hours
Security	Loss of access	4 Hours
Operational	A Notice of Violation (NOV) issued by NRC in accordance with 10CFR 2.201. Any potential violation identified that is beyond "green"	24 Hours
Operational	An event or condition that causes an NRC performance indicator to cross into "white, yellow or red"	24 Hours
Safety	Environmental, equipment or property damage, or disappearance of property, estimated to exceed $100,000	24 Hours

Attachment 3.1
Page 1 of 2

MATRIX OF EVENTS AND NOTIFICATION REQUIREMENTS

Category	Event	Time to Notify
HR	Employee termination for cause	24 Hours
HR	NSIP complaint - Discrimination	24 Hour
Legal	New lawsuit	24 Hours
Administrative	Public record request – Personnel	24 Hours
Business	Confirmed new Business GE $100K	24 hours
HR	NSIP complaint - All Others	Bi-weekly
Administrative	Public record request – Others	Bi-weekly

Attachment 3.1
Page 2 of 2

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

LCO 3.0.4 (continued)

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

1.7 PLANT SYSTEMS

1.7.1 Area Temperature Monitoring

RFO 1.7.1 Area temperatures shall be maintained within limits as shown in Table 1.7.1-1.

APPLICABILITY: When equipment in a room or area listed in Table 1.7.1-1 is required to be OPERABLE.

COMPENSATORY MEASURES

------------------------------NOTE------------------------------
Separate condition entry is allowed for each area.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. With one or more areas not within limits of Table 1.7.1-1.	A.1 Enter the condition referenced in Table 1.7.1-1. <u>AND</u> A.2 Initiate a Condition Report (CR).	Immediately 24 hours
B. As required by Compensatory Measure A.1 and referenced in Table 1.7.1-1.	B.1 Initiate action to restore area or room temperature to be within the Condition B limits of Table 1.7.1-1. <u>AND</u> B.2 Perform SR 1.7.1.1 for affected areas.	Immediately Once per 4 hours

(continued)

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
C. As required by Compensatory Measure A.1 and referenced in Table 1.7.1-1.	C.1 Restore area or room temperature to be within the limits of Table 1.7.1-1.	1 hour
D. Required action and associated Completion Time of Condition C not met.	D.1 Declare affected equipment as listed in Table 1.7.1-2 inoperable or associated LCO not met and enter the required action.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 1.7.1.1	Verify temperatures in rooms/areas listed in Table 1.7.1-1 are within limits.	As noted in Table 1.7.1-1

Table 1.7.1-1 (page 1 of 8)
Area/Room Temperature Limits

-----------------------------------NOTE-----------------------------------
When the area/room temperature is above the Condition B limit solely due to performance of required surveillances, when swapping units, or changing modes on area/room HVAC equipment, entry into the associated Conditions and Required Compensatory Measures may be delayed for up to 4 hours provided the associated function remains OPERABLE.

Description	Room/Area	Condition B Temp Limits	Condition C Temp Limits	Surveillance Frequency
Main Control Room(1)	C414	≤ 78°F	≤ 104°F	12 hours
DG Engine/Electrical Rooms				
HPCS DG3 Engine Room(1)	D100	≤ 112°F	≤ 122°F	12 hours
DG1 Engine Room(1)	D107	≤ 120°F	≤ 130°F	12 hours
DG2 Engine Room(1)	D110	≤ 120°F	≤ 130°F	12 hours
HPCS DG3 Elec Equip Room(1)(9)(18)	D114	≤ 104°F	≤ 111°F/ ≤ 120°F/ ≤ 124°F/ ≤ 129°F/ ≤ 141°F	12 hours
DG1 Elec Equip Room(1)	D115	≤ 104°F	≤ 129°F	12 hours
DG2 Elec Equip Room(1)	D116	≤ 104°F	≤ 129°F	12 hours
DG Support Areas/Rooms				
DG1 Storage Tank/Transfer Room	D101	≤ 104°F	≤ 142°F	none
DG2 Storage Tank/Transfer Room	D102	≤ 104°F	≤ 142°F	none

(continued)

(1) Monitor local temperature in room.
(9) See Table 1.7.1-2 for applicability to equipment vs temperature limit.
(18) See also HPCS DG Battery Room D114 on page 1.7.1-6.

Table 1.7.1-1 (page 6 of 8)
Area/Room Temperature Limits

Description	Room/Area	Condition B Temp Limits	Condition C Temp Limits	Surveillance Frequency
Reactor Bldg Support Areas/Rooms (continued)				
471' Open Areas (not elsewhere listed)(9)(10)	N/A	≤ 94°F	≤ 104°F	12 hours
441' Railway Bay(1)	R105	≤ 104°F	≤ 137°F	12 hours(17)
501' Open Areas/Rooms (not elsewhere listed)	N/A	≤ 94°F	≤ 104°F	none
522' Open Areas/Rooms (not elsewhere listed)(4)	N/A	≤ 100°F	≤ 104°F	31 days(8)
Fuel Pool Heat Exchanger Room	R506	≤ 104°F	N/A	none
548' Open Areas/Rooms (not elsewhere listed)(5)	N/A	≤ 94°F	≤ 104°F	31 days(8)
572' Open Areas/Rooms (not elsewhere listed)(6)	N/A	≤ 94°F	≤ 104°F	31 days(8)

(continued)

(1) Monitor local temperature in room.
(4) Monitor temperature for this area at a local point at 522' NW side.
(5) Monitor temperature for this area at two local points - 548' NW side and 548' S near SLC.
(6) Monitor temperature for this area at a local point at 572' N near SGT.
(8) Increase surveillance frequency to once per 12 hours if HVAC for this area/room is secured/inoperable.
(9) See Table 1.7.1-2 for applicability to equipment vs temperature limit.
(10) Monitor temperature for this area at a local point at 471' W near E-SH-9 or 10.
(17) Surveillance not required in MODE 4 or 5.

Table 1.7.1-2 (page 5 of 6)
Equipment Operability List

Area/ Room	Function	Limiting Temp	Affected EPN's	Ref	LCO/RFO
R15	RCIC Pump Room	150°F	RCIC-P-1 RCIC-DT-1	6	3.5.3
R105	441' RR Bay	137°F	CIA N2 Supply	5	3.5.1
R212	471' DC MCC Room	129°F	E-MC-S2/1A	3	3.8.7/3.8.8
R410	522' MCC Room Div II	129°F	E-MC-8B E-MC-8BA	3	3.8.7/3.8.8
R411	522' MCC Room Div I	129°F	E-MC-7B E-MC-7BA	3	3.8.7/3.8.8
R611	Hydrogen Recombiner Room Div I	104°F	CAC DIV I SGT DIV I	7	3.6.3.1 3.6.4.3
		129°F	E-MC-7BB	3	3.8.7/3.8.8
R612	Hydrogen Recombiner Room Div II	104°F	CAC DIV II SGT DIV II	7	3.6.3.1 3.6.4.3
		129°F	E-MC-8BB	3	3.8.7/3.8.8
	RB 471' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3
	RB 501' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3
	RB 522' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3
	RB 548' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3
	RB 572' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3
	RB 606' Open areas	104°F	SGT DIV I SGT DIV II	11	3.6.4.3

(continued)

Table 1.7.1-2 (page 6 of 6)
Equipment Operability List

Area/ Room	Function	Limiting Temp	Affected EPN's	Ref	LCO/RFO
PH A	SW Pump House A	122°F	SW-M-P/1A E-TR-7AF/1	1	3.7.1
		140°F	HPCS-M-P/2	1	3.5.1/3.5.2
PH B	SW Pump House B	122°F	SW-M-P/1B E-TR-8AF/1	1	3.7.1
N/A	Drywell	200°F	see Table 1.7.1-1 note 14	8	Operations to determine
N/A	Suppression Pool Air Space	200°F	see Table 1.7.1-1 note 14	12	Operations to determine
N/A	Area Under RPV	200°F	see Table 1.7.1-1 note 14	9	Operations to determine
N/A	Main Steam Tunnel	200°F	see Table 1.7.1-1 note 15	10	Operations to determine

References for Table 1.7.1-2

1. QID 829213
2. QID 252002
3. Calculation EQ-02-92-10
4. FSAR 3.11-1
5. PER 200-0060
6. QID 213032
7. Engineering Technical Memorandum TM-2123
8. QID 297009
9. QID 067005
10. QID 315025
11. Calculation NE-02-94-71
12. QID 195013
13. QID 184003
14. QID 063002

1.9 REFUELING OPERATIONS

1.9.2 Crane Travel

RFO 1.9.2 Crane travel with loads over fuel assemblies stored in the spent fuel storage pool racks shall be within the limits of Figure 1.9.2-1.

APPLICABILITY: With irradiated fuel stored in the spent fuel storage pool (SFP) racks.

COMPENSATORY MEASURES

------------------------------------NOTE------------------------------------
RFO 1.0.3 is not applicable.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirements of RFO not met.	A.1 Initiate actions to move the crane load from over the spent fuel storage pool racks.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 1.9.2.1	-------------------NOTE------------------- Only required when crane is in use. -- Perform system functional test.	7 days

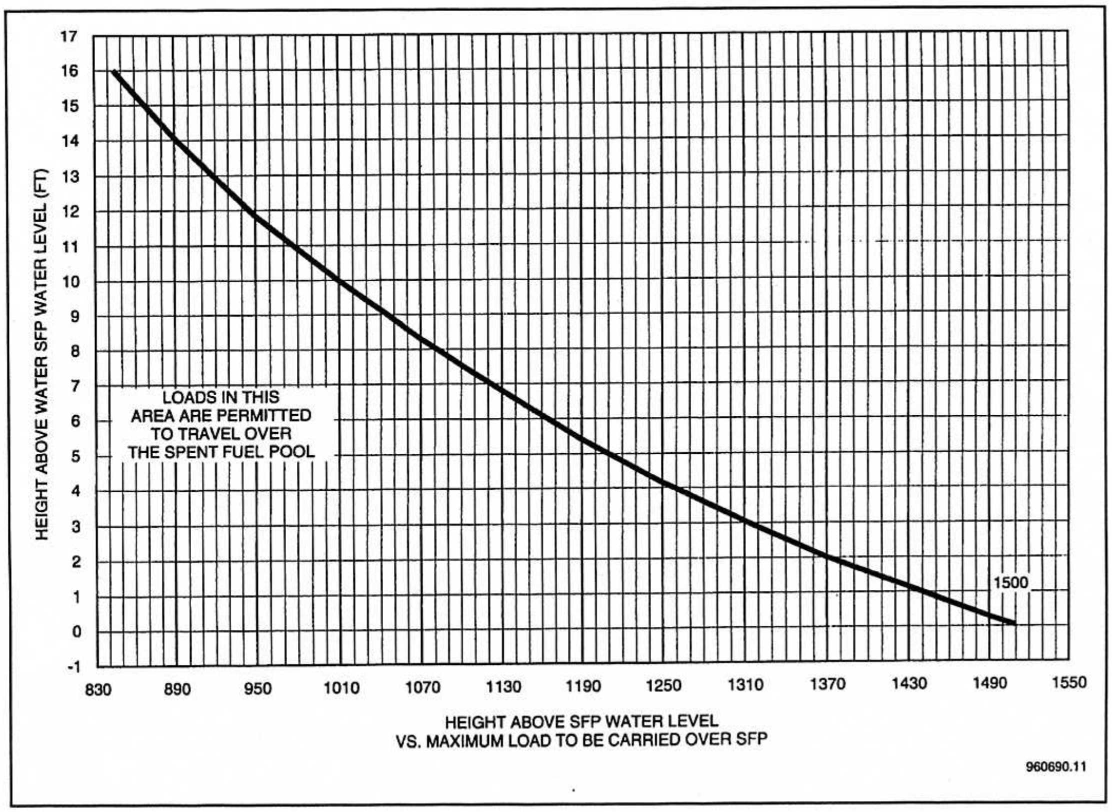

Figure 1.9.2-1
Crane Travel

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. OR	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.1 Verify $\geq$ 12 rods withdrawn. OR C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last calendar year. AND C.2.2 Verify movement of control rods is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	Immediately Immediately During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with BPWS by a second licensed operator or other qualified member of the technical staff.	During control rod movement

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-------------------NOTE------------------- Not required to be performed until 1 hour after any control rod is withdrawn at $\leq$ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	-------------------NOTE------------------- Not required to be performed until 1 hour after THERMAL POWER is $\leq$ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RBM is not bypassed: a. When THERMAL POWER is $\geq$ 30% RTP; and b. When a peripheral control rod is not selected.	92 days
SR 3.3.2.1.5	-------------------NOTE------------------- Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	92 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months
SR 3.3.2.1.7	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	 24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1 Restore control room AC subsystem to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1 --------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. -------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 3.a and 3.b. --------------------- Declare High Pressure Core Spray (HPCS) System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.d, 1.e, 1.f, 2.c, 2.d, 2.e, and 2.f. --------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	--------NOTE--------- Only applicable if HPCS pump suction is not aligned to the suppression pool. ---------------------	
		Declare HPCS System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>		
	D.2.1	Place channel in trip.	24 hours
	<u>OR</u>		
	D.2.2	Align the HPCS pump suction to the suppression pool.	24 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.g, 1.h, and 2.g. --------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. (continued)	F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCS or reactor core isolation cooling (RCIC) inoperable AND 8 days
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1	--------NOTE--------- Only applicable for Functions 4.b, 4.d, 4.e, 5.b, and 5.d. --------------------- Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued)	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCS or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3, 4[(a)],5[(a)]	2[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ -142.3 inches
b. Drywell Pressure – High	1,2,3	2[(b)]	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≤ 1.88 psig.
c. LPCS Pump Start – LOCA Time Delay Relay	1,2,3, 4[(a)],5[(a)]	1[(e)]	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 8.53 seconds and ≤ 10.64 seconds
d. LPCI Pump A Start – LOCA Time Delay Relay	1,2,3, 4[(a)],5[(a)]	1[(e)]	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 17.24 seconds and ≤ 21.53 seconds
e. LPCI Pump A Start – LOCA/LOOP Time Delay Relay	1,2,3, 4[(a)],5[(a)]	1	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 3.04 seconds and ≤ 6.00 seconds
f. Reactor Vessel Pressure – Low (Injection Permissive)	1,2,3	1 per valve	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 448 psig and ≤ 492 psig
	4[(a)],5[(a)]	1 per valve	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 448 psig and ≤ 492 psig
g. LPCS Pump Discharge Flow – Low (Minimum Flow)	1,2,3, 4[(a)],5[(a)]	1	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 668 gpm and ≤ 1067 gpm
h. LPCI Pump A Discharge Flow – Low (Minimum Flow)	1,2,3, 4[(a)],5[(a)]	1	E	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 605 gpm and ≤ 984 gpm
i. Manual Initiation	1,2,3, 4[(a)],5[(a)]	2	C	SR 3.3.5.1.6	NA

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated diesel generator (DG).

(e) Also supports OPERABILITY of 230 kV offsite power circuit pursuant to LCO 3.8.1 and LCO 3.8.2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

LCO 3.4.2 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. ---	
	Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Recirculation loop drive flow versus recirculation pump speed differs by ≤ 10% from established patterns. b. Recirculation loop drive flow versus total core flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns, or each jet pump flow differs by ≤ 10% from established patterns.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) – ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required SRVs inoperable.	A.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1165 ± 34.9 4 — 1175 ± 35.2 4 — 1185 ± 35.5 4 — 1195 ± 35.8 4 — 1205 ± 36.1	In accordance with the Inservice Testing Program
SR 3.4.3.2	Verify each required SRV opens when manually actuated.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (SRVs) – < 25% RTP

LCO 3.4.4 The safety function of four SRVs shall be OPERABLE.

APPLICABILITY: MODE 1 with THERMAL POWER < 25% RTP,
MODES 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1165 ± 34.9 4 — 1175 ± 35.2 4 — 1185 ± 35.5 4 — 1195 ± 35.8 4 — 1205 ± 36.1	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.2	-------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each required SRV opens when manually actuated.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------- LCO 3.0.3 is not applicable. ----------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	--------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SGT filter cooling recirculation valve can be opened and the fan started.	24 months

3.7 PLANT SYSTEMS

3.7.1 Standby Service Water (SW) System and Ultimate Heat Sink (UHS)

LCO 3.7.1 Division 1 and 2 SW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Average sediment depth in one or both spray ponds $\geq$ 0.5 ft and < 1.0 ft.	A.1	Restore average sediment depth to within limits.	30 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One SW subsystem inoperable.	B.1 --------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," for diesel generator made inoperable by SW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown," for RHR shutdown cooling subsystem made inoperable by SW System. --------------------- Restore SW subsystem to OPERABLE status.	72 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR Both SW subsystems inoperable. OR UHS inoperable for reasons other than Condition A.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify the water level of each UHS spray pond is $\geq$ 432 ft 9 inches mean sea level.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.2	Verify the average water temperature of each UHS spray pond is ≤ 77°F.	24 hours
SR 3.7.1.3	------------------NOTE------------------ Isolation of flow to individual components does not render SW subsystem inoperable. --- Verify each SW subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.1.4	Verify average sediment depth in each UHS spray pond is < 0.5 ft.	92 days
SR 3.7.1.5	Verify each SW subsystem actuates on an actual or simulated initiation signal.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 6.2.2.6.1	Verify the containment is aligned for VENTING or PURGING through the SGT system or the Primary Containment VENT and PURGE System.	Within 4 hours prior to the start and once per 12 hours during PURGING or VENTING

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two control room AC subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room AC subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. -----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC portion of the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System (hereafter referred to as the Control Room AC System) provides temperature control for the control room following isolation of the control room (from the normal intake and exhaust).

The Control Room AC System consists of two independent, redundant subsystems that provide cooling of recirculated control room air. Each subsystem consists of an air filter, two cooling coils (one normal and one emergency), a control room recirculation fan, ductwork, dampers, and instrumentation and controls to provide for control room temperature control. While there are two cooling coils, only the emergency cooling coil is required by this LCO. The emergency cooling coils are cooled by either the Emergency Chilled Water System, which consists of two chillers and two pumps (one chiller and pump combination for each emergency cooling coil) or by the Standby Service Water (SW) System. The SW System also provides cooling to the Emergency Chilled Water System chillers.

The Control Room AC System is designed to provide a controlled environment under both normal (using the non-safety related normal cooling coils) and accident (using the safety related emergency cooling coils) conditions. A single subsystem provides the required temperature control to maintain a suitable control room environment with a sustained occupancy of 10 persons. The design condition for the control room environment is 85°F when the emergency cooling coil is cooled by the Emergency Chilled Water System and 104°F when the emergency cooling coil is cooled by the SW System. The Control Room AC System operation in maintaining the control room temperature is discussed in the FSAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single active failure of a component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control when the emergency cooling coils are cooled by the Emergency Chilled Water System. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY. The Control Room AC System satisfies Criterion 3 of Reference 3.
LCO	Two independent and redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits. The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the emergency cooling coils (either cooled by the Emergency Chilled Water System or the SW System), control room recirculation fans, Emergency Chilled Water System chillers and pumps (if the Emergency Chilled Water System is being credited with providing cooling to the emergency cooling coils), ductwork, dampers, and associated instrumentation and controls. In addition, during conditions in MODES other than MODES 1, 2, and 3 when the Control Room AC System is required to be OPERABLE (e.g., during CORE ALTERATIONS), the necessary portions of the SW System and the ultimate heat sink are part of the OPERABILITY requirements covered by this LCO.

(continued)

BASES (continued)

APPLICABILITY

---------------------------NOTE---------------------------
Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate cooling methods.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room AC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2, and C.2.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall

(continued)

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

D.1

If both control room AC subsystems are inoperable in MODE 1, 2, or 3, the Control Room AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs with two control room AC subsystems inoperable, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe

(continued)

BASES

ACTIONS	E.1, E.2, and E.3 (continued) position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.
SURVEILLANCE REQUIREMENTS	SR 3.7.4.1 This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.
REFERENCES	1. FSAR, Section 6.4. 2. FSAR, Section 9.4.1. 3. 10 CFR 50.36(c)(2)(ii).

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with stored fuel oil level: 1. For DG-1 or DG-2, < 55,500 gal and ≥ 47,520 gal; and 2. For DG-3, < 33,000 gal and ≥ 28,340 gal.	A.1 Restore stored fuel oil level to within limit.	48 hours
B. One or more DGs with lube oil inventory: 1. For DG-1 or DG-2, < 330 gal and ≥ 283 gal; and 2. For DG-3, < 165 gal and ≥ 142 gal.	B.1 Restore lube oil inventory to within limit.	48 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1 Restore stored fuel oil total particulates to within limit.	7 days
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with required starting air receiver pressure: 1. For DG-1 and DG-2, < 230 psig and ≥ 150 psig; and 2. For DG-3, < 223 psig and ≥ 150 psig.	E.1 Restore required starting air receiver pressure to within limit.	48 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> One or more DGs with stored diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains: a. ≥ 55,500 gal of fuel for DG-1 and DG-2; and b. ≥ 33,000 gal of fuel for DG-3.	31 days
SR 3.8.3.2	Verify lube oil inventory is: a. ≥ 330 gal for DG-1 and DG-2; and b. ≥ 165 gal for DG-3.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each required DG air start receiver pressure is: a. ≥ 230 psig for DG-1 and DG-2; and b. ≥ 223 psig for DG-3.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	92 days

6.1 INSTRUMENTATION

6.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

RFO 6.1.1.1 The radioactive liquid effluent monitoring instrumentation channels in Table 6.1.1-1 shall be OPERABLE.

APPLICABILITY: In accordance with Table 6.1.1-1.

COMPENSATORY MEASURES

----------NOTE----------
Separate condition entry is allowed for each channel.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One or more required radioactive liquid effluent monitoring instrumentation channels inoperable.	A.1 Enter the condition referenced in Table 6.1.1-1 for the channel.	Immediately
B. As required by Compensatory Measure A.1 and referenced in Table 6.1.1-1.	B.1 Perform SR 6.2.1.1.1 on two independent samples of the batch to be released. <u>AND</u>	Prior to radioactive liquid release through the radwaste effluent line
	B.2 Verify the associated release rate calculations and the discharge valve lineup using two qualified members of the technical staff. <u>AND</u>	Prior to radioactive liquid release through the radwaste effluent line (continued)

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
B. (continued)	B.3 Restore the channel to OPERABLE status.	30 days
C. As required by Compensatory Measure A.1 and referenced in Table 6.1.1-1.	C.1 Analyze a grab sample for radioactivity (beta or gamma) of the associated pathway. The LLD shall be ≤ 1E-7 μCi/ml.	Once per 12 hours
	<u>AND</u>	
	C.2 Restore the channel to OPERABLE status.	30 days
D. As required by Compensatory Measure A.1 and referenced in Table 6.1.1-1.	D.1 Estimate the flow rate through the associated pathway.	At the beginning of the release and once per 4 hours during releases through the associated line
	<u>AND</u>	
	D.2 Restore the channel to OPERABLE status.	30 days
E. Required Compensatory Measure B.3, C.2, or D.2 and associated Completion Time not met.	E.1 Prepare and submit, in the Radioactive Effluent Release Report, the reason the channel was not restored to OPERABLE status within 30 days.	Upon submittal of current calendar year Radioactive Effluent Release Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 6.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 6.1.1.2	Perform SOURCE CHECK.	Prior to each radioactive release
SR 6.1.1.3	Perform SOURCE CHECK.	31 days
SR 6.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 6.1.1.5	Perform CHANNEL CALIBRATION.	18 months

Table 6.1.1-1
Radioactive Liquid Effluent Monitoring Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM COMPENSATORY MEASURE A.1	SURVEILLANCE REQUIREMENTS	ALARM/ TRIP SETPOINT
1. Liquid Radwaste Effluent Line Gross Radioactivity Monitor	(a)	1	B	SR 6.1.1.1 SR 6.1.1.2 SR 6.1.1.4 SR 6.1.1.5	(b)
2. Deleted					
3. Turbine Service Water System Gross Radioactivity Monitor	(d)	1	C	SR 6.1.1.1 SR 6.1.1.3 SR 6.1.1.4 SR 6.1.1.5	8E-6 μCi/ml Cs-137
4. Standby Service Water Gross Radioactivity Monitor	(d)	1 per loop	C	SR 6.1.1.1 SR 6.1.1.3 SR 6.1.1.4 SR 6.1.1.5	8E-6 μCi/ml Cs-137
5. Liquid Radwaste Effluent Line Flow Rate Monitor	(a)	1	D	SR 6.1.1.1 SR 6.1.1.4 SR 6.1.1.5	(c)
6. Plant Discharge Blowdown Line Flow Rate Monitor	(a)	1	D	SR 6.1.1.1 SR 6.1.1.4 SR 6.1.1.5	(c)

(a) When radioactive effluents are being discharged through this pathway.
(b) The alarm/trip setpoint of the Liquid Radwaste Effluent Line Gross Radioactivity Monitor shall be set to ensure the limits of RFO 6.2.1.1 are not exceeded for each batch of radioactive liquid effluent released. The alarm/trip setpoint of this channel shall be determined and adjusted in accordance with the methodology and parameters described in the ODCM and plant procedures.
(c) No alarm setpoints are required for these record only instruments.
(d) When there is flow in the system identified by this function.

6.2 RADIOACTIVE EFFLUENTS

6.2.2 Gaseous Effluents

6.2.2.6 VENTING or PURGING

RFO 6.2.2.6 VENTING or PURGING of the primary containment shall be:

a. Through one functional-for-filtration Standby Gas Treatment (SGT) System train during MODES 1, 2, or 3 provided the other train is OPERABLE during the first 24 hours of any VENTING or PURGING operation; or

b. Through one or two functional-for-filtration SGT System train(s) when deinerting the containment in MODE 4 during the first 24 hours of any VENTING or PURGING operation; or

c. Through the Primary Containment VENT and PURGE System when not using SGT following the first 24 hours of any VENTING and PURGING operation.

APPLICABILITY: MODES 1, 2, and 3 when VENTING or PURGING and when deinerting the containment. Not applicable during containment depressurization following 10 CFR 50, Appendix J, Type A testing.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirements of 6.2.2.6 not met.	A.1 Suspend VENTING and PURGING of the containment.	Immediately