Examination Cover Sheet

Exam ID: 2002 RONRC **Total Points:** 100

Exam Date: 08/05/2002

Description: 2002 NRC RO Written Examination

Student Name: ______________________________ **Date**: __________ **Grade:** __________

Graded By: ______________________________ **Date:** __________

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Reviewed By Examinee: ______________________________

Question 1 17898 (1 point(s))

The plant is operating at 100% power with a normal electric plant lineup when an oil leak occurs on the Normal Tranformer. The oil catches on fire. The fire brigide responds and attempt to extinguish the blaze. The oil and the fire is contained to the Normal Transformer area.

What is the potential effect of this fire?

a. The loss of one (1) Recirculation pump **only**.

b. A main generator trip AND Reactor scram **only**.

c. The loss of **BOTH** Recirculation pumps AND main generator trip AND Reactor scram.

d. The loss of **only** one (1) Recirculation pump AND main generator trip AND Reactor scram.

Answer 1

d. The loss of **only** one (1) Recirculation pump AND main generator trip AND Reactor scram.

The NORMAL transformer will trip eventually due to the fire.

REFERENCE: **C-2/A-8,** NORMAL TRANSFORMER LOCKOUT, COR0010102

Foils:

a. The main generator will trip as well.
b. A recirc pump will also trip.
c. Only one recirc pump will trip.

New
Difficulty 3
Cognitive Level 3

Enabling Objectives

INT0320134E0E0100 Given plant condition(s), determine from memory any automatic actions listed in the applicable Abnormal/Emergency Procedure(s) which will occur due to the event(s).

COR0010102001080A   Predict the consequences of the following on plant operation: Loss of Normal and Startup transformers

Skills

600000.AA2.04 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: The fire's extent of potential operational damage to plant equipment (2.8/3.1)

Question 2   17868   (1 point(s))

What is the effect of losing 125 VDC to the turbine generator?

a. Loss of DEH pressure.

b. Loss of main turbine remote trip capability.

c. DEH shifts to the standby pressure regulator.

d. The generator field breaker automatically opens.

Answer 2

b. Loss of main turbine remote trip capability.

Reference: 5.3DC125

Foils:

a. DEH pressure is not affected. The pumps are AC powered.
c. They are not affected by the power loss.
d. The generator field breaker must be manually opened.

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

COR0020702001080C   Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Systems with DC components (i.e., valves, motors, solenoids, etc.)

COR0011302001030E   Describe the interrelationships between Main Turbine Generator and Auxiliaries and the following: DC electrical distribution

Skills

245000.K1.09 Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) D. C . electrical distribution (2.7/2.7)

Question 3   17847   (1 point(s))

Following a a reactor scram from 100% power the main turbine generator has automatically tripped.

Which one of the following groups of breakers will the generator trip automatically open?

The main generator output breakers, the generator field breaker and the feeder breakers from . . .

a. the auto transformer.

b. the startup transformer.

c. the normal transformer.

d. busses 1A & 1B to busses 1F & 1G.

Answer 3

c. the normal transformer.

Reference: COR001-13-01, Section III Instrumentation and Controls.

Foils:

a. The generator field breaker is tripped and the startup breakers close vice open.
b. The generator field breaker is tripped.
c. The feeder breakers from the startup transformer will close vice open.

New
Difficulty: 2
Cognitive level:1

Enabling Objectives

COR0010102001130A   Predict the consequences of the following events on the AC Electrical Distribution System: Turbine/generator trip

Skills

295005.AA1.07 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.7 / 45.6) A.C. electrical distribution (3.3/3.3)

Question 4 17889 (1 point(s))

Why is Recirculation pump speed limited following a reactor scram?

a. To protect the Reactor Recirculation pump from damage due to lack of NPSH.

b. To prevent damage to the Recirculation pump thrust bearing due to inadequate film layer.

c. To protect the Reactor Recirculation pump motor from overload due to fluid density increase.

d. To prevent damage to the Reactor Recirculation pump seals caused by excessive differential pressure.

Answer 4

a. To protect the Reactor Recirculation pump from damage due to lack of NPSH.

REFERENCE: COR0022202

Foils:

b. Thrust bearing damage is not a reason for the #1 limiter.
c. Motor overload is not a reason for the #1 limiter.
d. Seal damage is a concern for the #1 limiter only if the Discharge valve is closed.

New
Difficulty: 2
Cognitive level:1

Enabling Objectives

COR0022202001100A Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following: Adequate Recirculation Pump NPSH

Skills

295006.AK3.06 Knowledge of the reasons for the following responses as they apply to SCRAM: (CFR: 41.5 / 45.6) Recirculation pump speed reduction: Plant-Specific (3.2/3.3)

Question 5   17859   (1 point(s))

The plant is operating at 30% when the reactor operator is directed to raise power using Recirculation flow. As the controller output is raised, a momentary (1.5 seconds) loss of signal occurs from the "A" Recirculation Flow Controller. The operator continues to raise the controller output for several more seconds.

How will the "A" Recirculation MG Set be affected by this momentary loss and operator action?

a.	Automatically run back to ~ 22% speed

b.	A scoop tube lockup will prevent any further speed change.

c.	After a 1.5 second pause recirculation pump speed will rise as directed.

d.	Speed will initially rise, then lower rapidly for 1.5 seconds then rise again for several seconds.

Answer 5

b.	A scoop tube lockup will prevent any further speed change.

Reference: 2.2.68, Attachment 1 Section 2.1.6

Foils:
a.	There is no runback.
b.	Scoop tube lockup prevents any speed changes.
d.	Scoop tube lockup prevents any speed changes.

New
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0022202001130F   Given plant conditions, determine if any of the following should occur: Recirculation MG set scoop tube lock.

COR0022201001060A   Given plant and/or reactor recirculation system conditions, apply the design features and/or interlocks that provide for the following: MG Set Scoop Tube Lockout

SKL012422200A030I Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: RR pump speed

Skills

202002.K1.05 Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Recirculation MG set: Plant-Specific (3.5/3.5)

Question 6   16478   (1 point(s))

The plant is at 20% power when the following annunciator alarms:

- 9-4-3 / A-3, RECIRC PUMP A SEAL TROUBLE
- RONAN (1820): RECIRC PUMP A SEAL STAGING FLOW HIGH

The following indications are observed:

- Seal cavity #1 (inner) pressure 950 psig
- Seal cavity #2 (outer) pressure 950 psig

What is the state of the A Recirculation Pump seals?

a. **Only** seal #1 is failed.

b. **Only** seal #2 is failed.

c. **Both** #1 and #2 seals are failed.

d. The seal staging flow orifice is clogged.

Answer 6

a. Only seal #1 is failed.

Equal pressure between RR-PI-32A (RR-PI-32B)SEAL 2 BRG CAV and RR-PI-33A (RR-PI-33B) SEAL 1 BRG CAV indicates a #1 seal failure.
Normally seal cavity 2 pressure should be about 1/2 of cavity 1 pressure.

REEFERENCE: 2.3_9.4.3
2.4RR; Attachment 3 - 1.3

Foils:

b. Very low pressure in seal cavity #2, indicated on RR-PI-32A (RR-PI-32B) SEAL 2 BRG CAV, indicates #2 seal failure. Upper cavity pressure will approach lower cavity pressure.
c. The pressure relationship between the seals would remain (1/2 for pressure). Would also receive (1819) Recirc Pump A Outer Seal Leakage Flow High Alarm.
d. Will not receive seal staging flow high alarm for this condition.

Bank
Difficulty 2
Cognitve Level 2

Enabling Objectives

COR0022202001140A Given plant conditions determine if the following has occurred: #1 seal failure only.

Skills

202001.A4.11 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Seal pressures: Plant-Specific (3.2/3.3)

Question 7   5497   (1 point(s))

Given the following conditions:

- The generator has been synchronized to the grid
- Bypass Valves are closed
- The load demand remains at 200
- Reactor power is raised using reactor recirculation flow

How will the bypass valves and DEH respond?

As reactor power is raised, turbine bypass valves . . .

a. remain closed **and** DEH remains in MODE 3.

b. remain closed **and** DEH remains in MODE 4.

c. open when load exceeds 200 MW **and** DEH transfers to MODE 3.

d. open when load exceeds 200 MW **but** DEH remains in MODE 4.

Answer 7

c. open when load exceeds 200 MW **and** DEH transfers to MODE 3.

EXPLANATION OF ANSWER: The load control signal enters a low value gate with pressure control signal. Whichever is lower will be passed through. Until megawatts exceeds 200 the load signal will be lower than the pressure control signal. At this point the opening of the control valves will be limited. This will cause the bypass valves to open. When the bypass valves open DEH will transfer to MODE 3.

REFERENCES: COR002-09-02, DEH. PR 2.2.77.1, DEH Control System

Foils:
a. Bypass valves will open and DEH was in Mode 4.
b. Bypass valves will open and DEH transfers to Mode 3.
d. DEH will transfer to Mode 3.

Bank
Difficulty: 2

Cognitive Level: 2

Enabling Objectives

COR0020902001040L   Describe how the DEH control system operates to control the following: Bypass valve position

COR00209020011100   Define the four DEH Control system operating Modes used at CNS.

COR00209020011200   Given a specific DEH Control system operating Mode, identify the parameter controlled and components positioned for controlling it.

Skills

295007.AK2.01 Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: (CFR: 41.7 / 45.8) Reactor/turbine pressure regulating system (3.5/3.7)

Question 8   17890   (1 point(s))

During a plant transient, SRV "D" lifts in response to high pressure. When a tailpipe vacuum breaker operates to equalize pressure, it sticks in the **OPEN** position.

What are the possible consequences of this failure of the SRV tailpipe vacuum breaker?

If the SRV lifts again, . . .

a. the pressure trapped in the discharge pipe would raise the operating setpoint of the SRV, causing possible damage to the Reactor Vessel and piping.

b. the water that was drawn up into the discharge pipe would cause a water hammer to occur, causing possible damage to the SRV, piping, and torus.

c. the steam passing through the SRV will be released directly into the torus airspace, bypassing the pressure suppression function of the primary containment.

d. the steam passing through the SRV will be released directly to the drywell atmosphere, producing primary containment conditions similar to a small break LOCA.

Answer 8

d. the steam passing through the SRV will be released directly to the drywell atmosphere, producing primary containment conditions similar to a small break LOCA.

REFERENCE: P&ID 2028

Foils:

a. This will not occur if a vacuum breaker sticks open.
b. This is the effect if the vacuum breakers stick closed.
c. The SRV tailpipe vacuum breakers relieve to the drywell, not torus airspace.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0021602001080H Predict the consequences a malfunction of the following would have on the NPR system: Discharge line vacuum breaker

Skills

239002.A2.01 Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45.6) Stuck open vacuum breakers (3.0/3.3)

Question 9   1499   (1 point(s))

An SRV is opened during normal power operations.

When read on MS-TR-166, SAFETY AND RELIEF VALVE LEAKAGE TEMPS, tail pipe temperature will indicate . . .

a. between 250°F and 280°F.

b. between 285°F and 325°F.

c. between 385°F and 400°F.

d. between 500°F and 550°F.

Answer 9

b. between 285°F and 325°F.

REFERENCE: COR002-16-02, page 23, section IV.E.2, rev. 09, Steam Tables

Foils:

a. Saturation temp for 30 psia.
c. Numbers to round out spacing of other numbers.
d. Saturation temp of vessel.

Bank
Difficulty 2
Cognitive Level 2

Provide to Candidate: Steam Tables

Enabling Objectives

SKL012421600A030F Given plant conditions, predict changes in the following NPR system components/parameters: SRV tailpipe temperatures

COR0021602001040A Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Tail pipe temperatures

Skills

218000.A4.06 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) ADS valve tail pipe temperature (3.5/3.6)

Question 10   12692   (1 point(s))

What is the reason for the decrease in MCPR associated with the "Feedwater Controller Failure (Maximum Demand)" transient?

a. Fuel heating due to loss of cooling

b. Fuel heating due to increased pressure

c. Positive reactivity due to void collapse

d. Positive reactivity from colder moderator

Answer 10

d. Positive reactivity from colder moderator

Reference: INT0060119 Student Text p. 33

Foils:
a.b.c. The loss of feedwater heating causes a reduction in moderator temperature and increases core inlet subcooling.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT00601190010400 Given an Anticipated Operational Transient and a list of reasons, select the correct response why the given transient would have MCPR limitations.

INT00601140010400 Given a transient and list of reasons, choose the reason the given transient would have MCPR limitations.

Skills

295014.AK2.06 Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) Moderator temperature. (3.4/3.5)

Question 11   17862   (1 point(s))

A station startup is in progress, reactor temperature is 178°F and rising at 80°F/hr. The Reactor Water Cleanup system (RWCU) is being used to control RPV water level when a high NRHX outlet temperature isolates RWCU.

How is RPV water level affected and what is the effect on the plant startup?

RPV water level . . .

a. rises. The startup may continue.

b. lowers. The startup may continue.

c. rises. The startup must be stopped due to inability to control RPV water level.

d. lowers. The startup must be stopped due to inability to control RPV water level.

Answer 11

c. rises. The startup must be stopped due to inability to control RPV water level.

Reference: 2.4RXLVL

Foils:
a.b.d. The loss of RWCU will prevent its use for RPV water level control (draining excess water via RWCU blowdown. Per 2.4RXLVL any power change must be stopped.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

SKL012412000A030A   Given plant conditions, predict changes in the following RWCU system components/parameters: Reactor vessel water level.

COR0012001001110D State how the following systems interrelate with the operation of the RWCU System: Main Condenser

Skills

204000.K3.02 Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor water level (3.1/3.1)

Question 12   17863   (1 point(s))

12 hours after shutdown, a loss of shutdown cooling has occurred. The crew has entered abnormal procedure 2.4SDC, "Shutdown Cooling Abnormal."

Which of the following systems can be used for alternate decay heat removal?

a. Fuel Pool Cooling

b. Reactor Water Cleanup

c. Turbine Equipment Cooling

d. Diesel Ventilation Cooling Towers

Answer 12

b. Reactor Water Cleanup

RWCU can be lined up per 2.2.66 for alternate decay heat removal.

Referernce: 2.4SDC

Foils:

a. Fuel pool cooling will not be available until the reactor head has been removed and the cavity flooded and the gates removed this will not occur within 12 hours of shutdown.
c. TEC is not an alternate decay heat removal system.
d. Diesel Ventilation Cooling Towers are not alternate decay heat removal systems.

New
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0012002001050C   Briefly describe RWCU operation under the following conditions: Loss of Shutdown Cooling

Skills

205000.K5.03   Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.5 / 45.3)
Heat removal mechanisms (2.8/3.1)

Question 13   5030   (1 point(s))

Given the following conditions:

- The Reactor Core Isolation Cooling (RCIC) System is operating in the Test Mode.
- The RCIC flow controller is in BAL.
- The RCIC TURB TEST Switch is in TURB TEST.
- The RCIC TURB TEST POWER Switch is ON.
- RCIC flow is 300 gpm and turbine speed is 2500 rpm (indicated in the control room).

The operator then jogs MO-30, RCIC Test Bypass To ECST Valve, in the CLOSED direction. MO-30 is still partially open.

What is the expected ***final*** response of RCIC turbine speed and system flow if all equipment operates as designed?

a. RCIC turbine speed rises.
   System flow rises.

b. RCIC turbine speed lowers.
   System flow lowers.

c. RCIC turbine speed is unchanged.
   System flow lowers.

d. RCIC turbine speed is unchanged.
   System flow rises.

Answer 13

c. RCIC turbine speed is unchanged.
   System flow lowers.

EXPLANATION OF ANSWER: With the Test Controls ON and no initiation signal present, the Flow Controller is isolated from the turbine governor and turbine speed is controlled with the Test Pot. Positioning MO-30 and adjusting turbine speed regulates flow. As MO-30 is jogged in the closed direction, flow will lower and turbine speed will remain constant.

REFERENCE: STCOR0021802 Section III.C
PR 2.2.67.1; Section 8.5
DWG 971E264 Sh. 5

Foils:

a. Speed remains constant, flow lowers
b. Speed remains constant.
d. Flow lowers.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0021802001100H Predict the consequences of the following on the RCIC system: System Valve closures and openings

Skills

217000.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: (CFR: 41.5 / 45.5) RCIC flow (3.7/3.7)

Question 14   17891   (1 point(s))

The plant was operating at full power with RCIC tagged out when it scrammed due to trip of both Feedwater pumps. HPCI injecting at full flow recovered RPV water level from - 50" (WR) to +35" (WR). HPCI was tripped as RPV water level reached + 35" (WR). Conditions are currently:

- Reactor pressure   700 psig (rising at 10 psig per minute)
- MSIVs   Open
- Time after scram   5 minutes
- CRD pumps   both tripped
- Drywell pressure   0.3 psig

If <u>no additional operator actions are taken</u>, what is the expected RPV water level response over the next 10 minutes and why?

RPV water level will . . .

a. lower below the low level alarm point due to shrink.

b. lower below the low level alarm point due to steam loads reducing RPV water inventory.

c. rise above the high RPV water level trip setpoint due to swell.

d. rise above the high RPV water level trip setpoint due to Startup Valve leakage exceeding decay heat requirements.

Answer 14

c. rise above the high RPV water level trip setpoint due to swell.

The specific volume change from ECST water to saturated liquid at 700 psig results in ~ 40% increase in gallons/inches of RPV level. 85" of cold water added = ~ 119" (34" more or + 69")

REFERENCE: Plant events, Steam Tables

Foils:

a. RPV water level will rise. Level will not shrink.

b. RPV water level will rise. BPVs will be closed at this pressure and remaining steam loads will not lower level under these conditions.

d. Rx pressure is above Condensate Booster pump discharge pressure.

New
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0021102001100F Predict the consequences of the following on the HPCI system: Low reactor water level

SKL012421100A030C Given plant conditions, predict changes in the following HPCI system components/parameters: Reactor water level

Skills

295008.AA2.05 Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Swell. (2.9/3.1)

Question 15   12444   (1 point(s))

The plant was operating at power when a loss of off-site power occurred. The reactor scrammed and HPCI started on low reactor water level. Reactor water level quickly recovered and the HPCI turbine tripped on high RPV water level. The following plant conditions were present:

- Reactor water level   45" (NR) (lowering slowly)
- Reactor pressure   850 psig (rising slowly)
- Drywell pressure   2.2 psig (rising slowly)

What is/are the **MINIMUM** action(s) required to restart HPCI ***at this time***?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

b. Momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

c. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

d. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

Answer 15

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

During the transient drywell pressure has risen to greater than the initiation setpoint for HPCI. Since an automatic initiation signal is present, if the operator depresses the Reactor Hi Water Level Signal Reset pushbutton the system will reinitiate.

REFERENCE: 2.2.33

Foils:
b, c, d - only the high level trip reset need be depressed.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0021102001120A   Given plant conditions, determine if the following HPCI actions should occur: System initiation

SKL012421100A030J   Given plant conditions, predict changes in the following HPCI system components/parameters: Turbine speed

COR0021102001100V   Predict the consequences of the following on the HPCI system: High reactor water level

Skills

206000.K4.07   Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Automatic system initiation: BWR-2,3,4 (4.3*/4.3)

Question 16   3790   (1 point(s))

Service Water is cross-connected to REC from the Control Room. SW is supplying cooling to the critical loops.

How are REC/Service Water systems affected (if at all) by a Group 6 isolation signal?

a. Service Water will be isolated.

b. Service Water **AND** REC will **NOT** be affected.

c. REC will automatically realign to the critical loops.

d. Some Service Water flow will bypass the critical loops.

Answer 16

d. Some Service Water flow will bypass the critical loops.

REFERENCES: SOP 2.2.65.1

Foils:

a. Service Water will NOT isolate.
b. Both Service Water and REC are affected.
c. REC will not automatically realign.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

SKL012421900A030L   Given plant conditions, predict changes in the following Reactor Equipment Cooling System components/parameters: SW/REC Crosstie

COR0021902001040B   Describe the REC design features and/or interlocks that provide for the following: Service Water Crosstie to REC

COR0021902001040D   Describe the REC design features and/or interlocks that provide for the following: Isolation of Non-Critical Cooling loops

COR0021902001110A   Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation

Skills

295018.AA1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.7 / 45.6) Backup systems (3.3/3.4)

Question 17   16475   (1 point(s))

With the plant at 65% power, an attempt is made to single **notch** withdraw a control rod from notch 24 to notch 26. While the control rod is being withdrawn, a malfunction of the RMCS timer causes a constant withdrawal signal to be sent to the selected control rod.

Assume NO additional operator actions and that all control rods move at nominal control rod speed.

What is the FINAL POSITION of the control rod?

a. Notch 26

b. Notch 28

c. Notch 30

d. Notch 48

Answer 17

b. Notch 28

Timer malfunction deselects the rod after 2 seconds (which is ½ second longer than the normal timer), causing the control rod to be de-selected

REFERENCE: IOP 4.3; Attachment 3, 1.3.2

Foils:
a. From the given power level an Rx scram will not occur.
c. An extra ½ second will not result in a two (2) notch change.
d. Rod will be de-selected after 2 seconds of motion.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0022002001040A   Describe the RMCS design features and/or interlocks that provide for the following: Detection of sequence timer malfunction

COR0022002001060A Given a RMCS control manipulation, predict and explain the response of the following: Control rod position

COR0022002001080A Given a specific RMCS and/or RPIS malfunction, determine the effect on any of the following: Ability to move control rods

Skills

201002.A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Control rod position (3.4/3.3)

Question 18   5006   (1 point(s))

Given the following conditions:

- A plant startup is in progress with power at 8%
- Group 42 rods are being withdrawn
- ONLY one rod remains to be withdrawn in Group 42
- All other control rods are at their proper position
- The operator attempts to select AND withdraw a rod in Group 43 one notch

What is the response of the Group 43 control rod?

The Control rod will . . .

a. **NOT** withdraw and withdrawal blocks will be applied.

b. **NOT** select and **NO** withdrawal blocks will be applied.

c. withdraw one notch, then withdrawal blocks will be applied.

d. withdraw one notch and **NO** withdrawal blocks will be applied.

Answer 18

d. withdraw one notch and **NO** withdrawal blocks will be applied.

EXPLANATION OF ANSWER: RWM will latch to group 43 when the first rod moves off of its insert limit. This will leave the group 42 rod as an insert error. No rod blocks will result from one insert error.

REFERENCE: PR 4.2 Rod Worth Minimizer

Foils:

a. Rod will withdraw and no blocks are applied.
b. Nothing prevents selecting the rod in this condition.
c. no rod blocks will result.

Bank

Difficulty 4
Cognitive Level 2

Enabling Objectives

COR0022602001040C Describe the RWM design features and/or interlocks that provide for the following: Withdraw blocks and errors

COR0022602001050I Briefly describe the following concepts as they apply to the RWM: Withdraw block

COR0022602001050C Briefly describe the following concepts as they apply to the RWM: Latched groups

COR0022602001080B Predict the consequences of the following items on the RWM: Out of sequence rod movement

Skills

201006.K4.01 Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Insert blocks/errors: P-Spec (Not-BWR6) (3.4/3.5)

Question 19 17865 (1 point(s))

A LOCA is in progress with drywell spray in service on the "A" RHR loop. "B" RHR loop is in torus cooling with suppression pool temperature 78°F. Due to a logic failure, the RHR drywell spray valves will not automatically close.

What is a possible consequence if the Reactor Building to Torus Vacuum Breakers fail CLOSED under these conditions?

a. Implosion of the primary containment.

b. Over-pressurization of the primary containment.

c. Excessive differential pressure from the Drywell to the Torus.

d. Excessive differential pressure from the Torus to the Drywell.

Answer 19

a. Implosion of the primary containment.

The Reactor Building to Torus Vacuum Breakers open on a negative pressure in the torus to protect against external over-pressurization which is caused by RHR condensing the steam in the torus free air space.

Reference: Tech. Spec Bases section B.3.6.1.7

Foils:

b. This failure would not result in containment overprssure.
c. The downcomers prevent this pressure exceeding ~ 2 psid.
d. Torus to drywell vacuum breakers would open on a high internal pressure.

New
Difficulty 2
Cognitive Level 3

Enabling Objectives

COR0022302001080L   Predict the consequences a malfunction of the following will have on the RHR system: Reactor building to Suppression chamber vacuum breakers

COR0020302001230D   Predict the consequences of a malfunction of the following on the Primary containment: Drywell vacuum relief.

COR0022302001040H   Describe the interrelationship between the RHR system and the following: Primary Containment

Skills

230000.K6.09   Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE: (CFR: 41.7 / 45.7) Reactor building to suppression pool vacuum breakers (3.5/3.8)

Question 20   9228   (1 point(s))

The plant was at 100% power when a large break LOCA occurred. Reactor water level initially lowered to +5" (corrected Fuel Zone), then was restored using RHR pumps which automatically started. To control the level rise, the BOP stopped RHR pumps A and D by momentarily placing the control switches to STOP.

Before any additional actions were taken, a loss of off-site power (LOOP) occurred.

Which RHR pumps will be operating ten (10) seconds after the DGs energize the emergency buses, if any?

a. All RHR pumps remain OFF.

b. All RHR pumps are running.

c. RHR pumps B and C are running. Other pumps remain OFF.

d. RHR pumps A and D are running. Other pumps remain OFF.

Answer 20

c. RHR pumps B and C are running. Other pumps remain OFF.

A and D RHR pumps are OFF with a STOP signal sealed in. Even if power is lost and restored the STOP signal for A and D RHR pumps remains sealed in. B and C RHR pumps will start 5 seconds after power is restored to their respective bus.

REFERENCE: 2.2.69; Attachment 1 - 2.1.1.2.a, 2.1.1.2.b, 2.1.2.2

Foils:

a. B and C pumps start after 5 seconds.
b. A and D RHR pumps remain OFF.
d. A and D pumps remain OFF. B and C RHR pumps will be running.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0022302001130D State the purpose of the following items related to the RHR system: Pumps

COR0022302001030F Describe RHR System design feature(s) and/or interlocks which provide for the following: Emergency Diesel Generator load sequencing

COR0022302001060I Given an RHR control manipulation, predict and explain changes in the following: Emergency Diesel Generator loading

COR0022302001150B Given plant conditions, determine if the following should occur: RHR pump start

Skills

203000.K2.01 Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (3.5*/3.5*)

Question 21   1729   (1 point(s))

The RPV is at 30 psig operating in Shutdown Cooling using the "B" RHR pump.

The SDC Outboard Isolation Valve, MO-17, will. . .

a. close if Reactor pressure is above 72 psig or a PCIS Group 2 isolation signal is received.

b. remain "as is" under all conditions, because it is maintained de-energized once Shutdown Cooling is established.

c. close if a PCIS Group 2 isolation signal is received, but will remain open if Reactor pressure rises above 72 psig.

d. remain "as is" under all conditions, because the Shutdown Cooling Isolation logic is disabled below 72 psig Reactor pressure.

Answer 21

a. close if Reactor pressure is above 72 psig **OR** a PCIS Group 2 isolation signal is received.

REFERENCE: RHR Text, Procedure 2.2.69, Attach. 1, step 2.2.6.3

Foils:

b. The valve is not allowed to be de-energized until Mode 4 is reached.
c. High Rx pressure will close the SDC isolation valve.
d. The isolation logic is never disabled, even in EOPs.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0022302001030B Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of piping over-pressurization

COR0022302001030J Interpret RHR System design feature(s) and/or interlocks which provide for the following: High pressure isolation

COR0022302001040L Describe the interrelationship between the RHR system and the following: PCIS

Skills

223002.K1.08 Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Shutdown cooling system/RHR (3/4/3.5)

Question 22   1744   (1 point(s))

Following a LOCA, the following conditions are present :

- Reactor pressure   700 psig
- RPV water level   - 100 in (wide range)
- Drywell pressure   11.0 psig
- Torus water temp   104 °F

What are the **MINIMUM** actions that are required in order to initiate Drywell Sprays with the "A" RHR loop? (NOTE: The choices are listed in MIMIMUM to MAXIMUM order.)

a. Place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

b. Depress Containment Spray Initiation Signal Reset pushbuttons, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

c. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

d. Place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

Answer 22

c. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

REFERENCE: RHR Text (COR0022302), Procedure 2.2.69, Attach. 1, Note page 43 & 2.2.10.2

Foils:

a. Containment Cooling Valve Control Permissive switches must be placed in MANUAL
b. There is no need to reset the containment spray initiation.
d. There is no need to override 2/3 core height.

Bank
Difficulty 2
cognitive Level 2

Enabling Objectives

COR0022302001030P Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling

COR0022302001170C Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Skills

226001.A1.05 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: (CFR: 41.5 / 45.5) System lineup (3.1/3.4)

Question 23   16499   (1 point(s))

During operation at full power annunciator 9-3-3/A-5, CORE SPRAY B BREAK DETECTION is received.

NO other annunciators alarm. A station operator sent to the d/p indicating switch reports that the d/p is +4.0 psid.

What is the significance of this alarm and d/p indication on core spray flow during a subsequent Core Spray initiation?

Core spray flow will flow …

a. into the Drywell through the broken pipe.

b. inside the core shroud and out the broken pipe.

c. into secondary containment through the broken pipe.

d. into the downcomer region of the reactor through the broken pipe.

Answer 23

d. into the downcomer region of the reactor through the broken pipe.

REFERENCE: 2.3_9-3-3, ST COR002-06-02

Justification: The alarm and d/p reading indicate the break is outside the shroud but inside the reactor.

Foils:

a. The indicated d/p would be pegged high (+1000 psig).
b. The indicated d/p would be low -3.5 psig.
c. The instrument measures d/p downstream of the check valve inside the primary containment.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0020602001090C   Predict the consequences of the following items on the Core Spray System:
Core Spray line break

Skills

209001.A4.11   Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
System flow (3.7/3.6)

Question 24   17871   (1 point(s))

The 12.5KV System has been removed from service. What Fire Protection System Fire Pumps (C, D, E) are available for use 15 minutes after the loss of 12.5KV?

a.   **Only** C

b.   **Only** D

c.   C **and** D

d.   D **and** E

Answer 24

c.   C **and** D

C Fire Pump is powered from MCC-E and the Diesel pump ("D") will start from its batteries.

Reference: 2.2.90: 6.3, 6.26,6.27, and 6.28, COR0010501

Foils:
a.   D will start as well
b.   C will start as well
d.   E pump has lost power

New
Difficulty 2
Cognitive level 1

Enabling Objectives

COR0010502001060A   State the electrical power supply to the following: Diesel Fire Pump starting circuitry

COR0010502001060B   State the electrical power supply to the following: Electric Fire Pumps "C" & "E"

Skills

286000.K2.02 Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (2.9/3.1)

Question 25 16470 (1 point(s))

Given the following conditions:

- The piping just downstream of the 'A' SBLC squib valve (SLC-14A) is completely obstructed.

- The control switch for SLC PUMP A on Panel 9-5 is taken to the START position, then to the STOP position when the no-flow condition is recognized.

- The control switch for SLC PUMP B on Panel 9-5 is taken to the START position. The SLC Pump B trips. Then the control switch is taken to the STOP position.

Which one of the following describes the **MINIMUM** steps necessary to inject SLC under the current conditions? (Note: The choices are listed in MINIMUM to MAXIMUM order.)

a. Return the control switch for SLC PUMP A on Panel 9-5 to the START position.

b. Place the control switches for SLC PUMP A **AND** SLC PUMP B on Panel 9-5 to START.

c. Start the B SLC Pump using the local test station keylock switch.

d. Perform alternate boron injection with any available system per EOP-5.8.8.

Answer 25

a. Return the control switch for SLC PUMP A on Panel 9-5 to the START position.

Justification: Either pump can discharge through either squib valve once both switches are placed in START. the 'B' squib valve fires when SLC PUMP B is started, when A is re-started it will discharge through the B squib valve.

REFERENCE: 2.2.74; Attachment 1.2.2, 1.3.1, 1.3.2, 1.3.3, 1.3.4

foils:

b. No need to place the SLC PUMP B control switch in start the squib valve has already been fired.
c. The breaker has tripped this would not start the pump.
d. No need to take the time for alternate injection. A SLC pump can inject.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0022902001060D   Briefly describe the following concepts as they apply to SLC system: Squib valve operation

COR0022902001080D   Given a SLC component manipulation, predict and explain the changes in the following: Flow indication

Skills

211000.K5.04   Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: (CFR: 41.5 / 45.3) Explosive valve operation (3.1/3.2)

Question 26   16625   (1 point(s))

The plant is at 100% power with TIP traces in progress. Only "A" TIP machine is being used at this time. Currently, TIP "A" has reached the Core Bottom Limit and is moving at slow speed to the Core Top Limit. The IN-CORE light is ON.

A reactor scram due to low RPV water level occurs. One (1) minute later, an operator observes:

- TIP valve indication on Containment Isolation display (Panel 9-3) is RED
- IN-SHIELD light for TIP "A" is ON at Panel 9-13
- Drywell pressure is normal

What action is ***required***?

a. Fire TIP "A" shear valve.

b. Close TIP "A" ball valve.

c. Manually retract TIP "A" to fire the shear valve.

d. Manually retract TIP "A" to close the ball valve.

Answer 26

b. Close TIP "A" ball valve.

The indications at Panel 9-3 indicate that a ball valve is open. If red light (Panel 9-3) stays on, at least one TIP ball valve has not closed. After automatic withdrawal of the TIP on the PCIS group 2 isolation signal, the ball valve failed to automatically close. This failed automatic action requires immediate operator action to manually perform the ball valve closure. The procedure directs the operator to attempt to manually retract TIP. Since the TIP is already retracted (IN-SHIELD light is on), this action is not necessary.
If ball valve cannot be closed and there are indications of a reactor coolant leak in drywell (as evidenced by the high drywell pressure) then fire appropriate shear valve by operating appropriate keylock switch.

Per 2.0.1.2, step 2.3 "Operators shall ensure automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals per their safety function design.

Upon recognition of a failure of automatic safety feature, Operators shall manually perform those actions necessary to fulfill the safety function."

REFERENCE: 4.1.4; Section 6 and Attach. 1, step 2.1.6, 2.0.1.2 step 2.3

Foils:

a. There is no indications of a LOCA and no attempt has yet been made to close the ball valve.
c. The TIP has already retracted and retracting the TIP does not fire the shear valve.
d. The TIP has already retracted.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0023102001110A Describe the TIP system design features and/or interlocks that provide for the following: Primary containment isolation

COR0023102001130C Given a TIP system control manipulation, predict and explain the changes in the following parameters: Valve status

COR0023102001140H Predict the consequences of the following on the TIP system: High primary containment pressure

COR0023102001160B Given plant conditions, determine if any of the following TIP actions should occur: Ball valve closure

Skills

215001.K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7) Primary containment isolation system: Mark-I&II (Not- BWR1) (3.1/3.4)

Question 27   1068   (1 point(s))
Given the following conditions:

- A Reactor Startup is in progress.
- Power is rising with a stable period.
- SRM detectors are withdrawn except for SRM "A" which fails to withdraw.
- The SRM UPSCALE OR INOPERATIVE alarm has been received.
- SRM "A" is **NOT** bypassed.

If power continues to rise, what is the FIRST point that control rods will be able to be withdrawn?

a. ALL IRMs are on Range 2.

b. ALL IRMs are on Range 3.

c. ALL IRMs are on Range 8.

d. The Mode switch is placed to RUN.

Answer 27

c. ALL IRMs are on Range 8.

IRMs on range 8 will bypass the SRM Rod Block.

Reference: COR0023002, 4.1.1

Foils:

a.b. SRM Rod Block is not bypased
d. Question asked for the FIRST point the control rods can be moved.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0023002001060F Describe the SRM system design features and/or interlocks that provide for the following: IRM/SRM interlock

Skills

215004.A3.04 Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status (3.6/3.6)

Question 28 1080 (1 point(s))

Given the following conditions:

- A plant startup is in progress.
- The Recirculation flow input signal to the Average Power Range Monitors (APRMs) is 50%.
- As Recirculation flow is raised, the output signal from the "B" Flow Unit remains at 50%.
- Actual Recirculation loop flows respond as expected.

As Recirculation flow continues to be raised, what will be the FIRST effect on plant operation from the APRMs/Flow Unit?

a. A half scram will occur due to flow biased neutron flux high.

b. A control rod block will occur due to a Flow Unit Comparator trip.

c. A control rod block will occur due to flow biased neutron flux high.

d. A half scram will occur due to a Flow Unit UPSCALE OR INOP trip.

Answer 28

b. A control rod block will occur due to a Flow Unit Comparator trip.

EXPLANATION OF ANSWER: A Flow Unit Comparator trip will occur when the A Flow Unit exceeds 60% (10% difference)

REFERENCES: STCOR002-01-02, page 16, section III.B, rev. 12.
2.3_9-5-1 9-5-1/A-4

Foils:

a. The margin to the SCRAM is large such that the Comparator trip will occur first.
c. The margin to the APRM rod block is large such that the Comparator trip will occur first.
d. Flow units do not directly cause scram signals.

Bank

Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0020102001050G   Describe the interrelationships between the Average Power Range Monitor System and the following: Flow converter/comparator network

Skills

215005.K1.16   Knowledge of the physical connections and/or cause- effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Flow converter/comparator network: Plant-Specific (3.3/3.4

Question 29   1120   (1 point(s))

Given the following conditions:

- The plant is performing a startup following a refueling outage
- The Reactor Mode Switch is in START & HOT STBY
- The "A" Reactor Protection System (RPS) MG set is being returned to service following maintenance
- Power is on IRM Range 3.
- When the operator placed the RPS Bus "A" Power Supply Selector to "NORMAL," a full reactor scram resulted (all rods in)

What additional condition would have caused this full reactor scram?

a. The Reactor Protection System shorting link switches are closed.

b. IRM Channel "E" was upscale and **NOT** bypassed during the transfer.

c. A Division 2 PCIS Group 1 half isolation was present during the transfer.

d. APRM Channel "F" was upscale and **NOT** bypassed during the transfer.

Answer 29

d. APRM Channel "F" was upscale and **NOT** bypassed during the transfer.

REFERENCES: STCOR002-01-02
PR 4.1.3, page 2, section 2.2.3, rev. 18.

Foils:

a. The shorting link switches are normally closed.
b. Channel E is fed by 24 VDC "A".
c. RMS in "Startup/Hot Standby" bypasses MSIV closure scram.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0020102001060A   State the electrical power supplies to the following: LPRM channels

COR0020102001060B   State the electrical power supplies to the following: APRM channels

Skills

215005.K2.02   Knowledge of electrical power supplies to the following: (CFR: 41.7) APRM channels (2.6/2.8)

Question 30   16433   (1 point(s))

A power ascension is in progress. Annunciator 9-5-1/D-4 RBM UPSCALE/INOP alarms.
At Panel 9-14, using the Meter Function Switch on RBM "A", the following indications are observed:

- AVERAGE   89%
- APRM SIG   78%
- COUNT   2.0 v (25 % of assigned inputs)
- BLOCK   94%

What is the status of RBM "A"?

a. RBM is inoperable from too few LPRM inputs.

b. RBM is operable, but an edge rod has been selected.

c. RBM is inoperable from a failure to null with the APRM.

d. RBM is operable, but local power is above the APRM signal.

Answer 30

a. RBM is inoperable from too few LPRM inputs.

Too few LPRM inputs count circuit shows less than 50%

REFERENCE: 2.3_9-5-1, 4.1.5; 1.2.4.1

Foils:

b. RBM is inoperable and edge rod would not cause the annunciator.
c. A failure to null has not occurred, there are too few LPRM inputs.
d. RBM is inoperable, local power is below APRM power but this is not the cause for a block, the RBM power is below the block setpoint.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0022402001130A Given plant conditions, determine if any of the following should occur: RBM control rod withdrawal block.

COR0022402001130C Given plant conditions, determine if any of the following should occur: RBM inop trip.

Skills

215002.A3.02 Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Meters and recorders: BWR-3,4,5 (3.1/3.0)

Question 31   16511   (1 point(s))

While operating steady state at 100% power the following indications are observed:

- Reactor power lowers
- Narrow Range reactor water level rises
- Indicated core plate d/p lowers
- Indicated TOTAL core flow rises
- "A" and "B" recirculation loop flows rise by the same amount

Which one of the following failures caused the above conditions?

a. A shroud support access hole cover has failed.

b. One (1) of the Jet pumps has a blocked throat.

c. One (1) recirculation pump's speed has raised to maximum.

d. Flow through a control cell (four fuel bundles) has been blocked.

Answer 31

a. A shroud support access hole cover has failed.

A shroud support access hole cover is ~ 19" in diameter. The effect to recirculation flow and core power by the separation of a cover could be significant. If a cover should separate, a flow path which bypasses the core is established which reduces the hydraulic resistance to flow through the core. This condition would indicate an increase in total core flow but actual flow through the core would drop and cause power to drop.

REFERENCES: 2.4RXPWR; 5.3, 6.2, 6.3

Foils:

b. Loop flows will lower in one loop and reactor water level change would not be discernible.
c. Would not provide these indications.
d. This would lower core flow.

Bank
Difficulty 3

Cognitive Level 1

Enabling Objectives

COR0011502001060C Given plant operating status, predict and explain the changes in the following parameters associated with the following: Reactor Power

COR0011502001060A Given a specific Nuclear Boiler system malfunction, determine the effect on any of the following: Reactor Water Level

COR0021502001020I Describe the interrelationships between NBI and the following: Reactor Vessel

Skills

290002.K3.07 Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: (CFR: 41.7 / 45.4) Nuclear boiler instrumentation (3.1/3.1)

Question 32   17860   (1 point(s))

The plant is operating at 50% power when a DEH malfunction causes the Main Turbine Bypass valves to rapidly open.

How is Wide Range RPV water level indication affected and why?

The __________ in flow resistance in the core region causes a/an _________ in the mass above the variable leg tap, causing a ________ in indicated RPV water level.

a. increase, increase, rise

b. increase, decrease, reduction

c. decrease, increase, rise

d. decrease, decrease, reduction

Answer 32

a. increase, increase, rise

Reference: COR002-15-02, 4.6.1, Attachment 2.

Foils:

b. The mass above the variable leg increases and indicated level rises.
c. The flow restriction rises.
d. The flow restriction rises, the mass above the variable leg increases and indicated level rises.

New
Difficulty 4
Cognitive Level 3

Enabling Objectives

COR0021502001040K Briefly describe the following concepts as they apply to NBI: Effects on level indication due to rapid changes in void fraction

Skills

216000.K5.12 Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.5 / 45.3) Effects on level indication due to rapid changes in void fraction (3.2/3.3)

Question 33   16435   (1 point(s))

The plant is operating at power with the following reactor vessel level control alignment:

- RFC-LC-83, MASTER LEVEL CONTROLLER in balance
- RFC-MA-84A, FW CONTROLLER STATION A in balance
- RFC-MA-84B, FW CONTROLLER STATION B in balance
- Feedwater flow is approximately $9.6x10^6$ lbm/hr.
- Steam flow is approximately $9.6x10^6$ lbm/hr.
- RPV water level is +35 inches.
- ALL listed controllers have been nulled.

The Master Controller OUTPUT slowly fails downscale. The operator places the "A" and "B" RFP controller mode switches in MANUAL when RPV water level lowers to +27 inches.

Assuming NO additional action is taken by the operator, what is the response of Feedwater Flow and RPV water level?

Feedwater flow . . .

a. rises to $9.6x10^6$ lbm/hr. Level rises to +42 inches.

b. rises to $9.6x10^6$ lbm/hr. Level remains at +27 inches.

c. rises above $9.6x10^6$ lbm/hr. Level rises to +35 inches.

d. remains below $9.6x10^6$ lbm/hr. Level continues to lower.

Answer 33

b. rises to $9.6x10^6$ lbm/hr. Level remains at +27 inches.

REFERENCE: 2.2.28; Attachment 1, Sections 1.2.5, 1.2.6 and 2.4RXLVL

Foils:

a. Level will not rise.

c. Feed flow will not rise above $9.6x10^6$ lbm/hr. Level will not rise. No recirc runback occurs.

d. Feed flow rises to $9.6x10^6$ lbm/hr. Level does not lower.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0023202001070A Given a RVLC system control manipulation, predict and explain the changes in the following parameters: RPV water level

COR0023202001070B Given a RVLC system control manipulation, predict and explain the changes in the following parameters: RFP Speed/Feed Flow

Skills

259002.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water level (3.8/3.8)

Question 34   1644   (1 point(s))

What effect (if any) will a high intake radiation have on the Main Control Room HVAC system?

a. There is no automatic response to this signal.

b. The system suction isolates and air is recirculated through the normal flowpath.

c. The system lineup shifts to process all outside air through the emergency filter train.

d. The system suction isolates and air is recirculated through the emergency filter train.

Answer 34

a. There is no automatic response to this signal.

(System mod removed intake monitor from initiation logic and replace it with Group 6 signal)

REFERENCES: COR0010802

Foils:

b. The system does not respond to high intake radiation anymore.
c. The system does not respond to high intake radiation anymore.
d. The system does not respond to high intake radiation anymore.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0010802001120A   Describe the control Room HVAC design features and interlocks that provide for the following: Control room HVAC reconfigurations

COR0010802001140A   Briefly describe the following concepts as they apply to Control Room HVAC: Airborne contamination (e.g., radiological,toxic gas, smoke) control

COR0010802001200A   Predict the consequences of the following items on the Control Room HVAC: Initiation/reconfiguration

Skills

295038.EA1.07 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) Control room ventilation: Plant-Specific. (3.6/3.8)

Question 35   16512   (1 point(s))

The plant is at 100% power. Reactor Building Ventilation is operating as follows:

- EF-1-RB, Exhaust Fan, running
- SF-R-1A-B, Supply Fan, running
- BF-R-1B, Exhaust Booster Fan, running
- ALL other fans are selected to STBY
- HV-DPIC-835A is in AUTO with reactor building pressure is -0.30" wg

How will the Reactor Building Ventilation system respond if the reactor building pressure degrades to -0.25" wg?

a. Supply Fan SF-R-1A-A starts and the supply fan vortex dampers position to correct the reactor building pressure.

b. Exhaust Fan EF-1-RA starts and the exhaust fan vortex dampers position to correct the reactor building pressure.

c. Operating fan lineup does NOT change. Supply fan vortex dampers position to correct the reactor building pressure.

d. Operating fan lineup does NOT change. Exhaust fan vortex dampers position to correct the reactor building pressure.

Answer 35

d. Operating fan lineup does NOT change. Exhaust fan vortex dampers position to correct the reactor building pressure.

Exhaust fan, exhaust booster fan, and supply fan will auto start when the control switch is in standby and Reactor Building pressure is below -0.15" wg and above -0.35" wg. This threshold has not been reached. The exhaust fan vortex dampers will position to improve the reactor building pressure.

REFERENCE: 2.2.47

Foils:

a. No fans auto start. Exhaust dampers reposition.

b. No fans auto start.
c. Exhaust dampers reposition.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0010802001110C Describe the HVAC design features and interlocks that provide for the following: Automatic starting and stopping of fans

COR0010802001130B Describe the control Room HVAC design features and interlocks that provide for the following: Differential pressure control

COR0010802001140B Briefly describe the following concepts as they apply to Control Room HVAC: Differential pressure control

Skills

295035.EA1.01 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.7 / 45.6) Secondary containment ventilation system (3.6/3.6)

Question 36   16496   (1 point(s))

The plant is at 100% power with the following conditions:

- Standby Gas Treatment (SGT) Exhaust Train A is being placed in service to support a surveillance test
- The control switch for SGT Fan 1F is in STBY
- The control switch for SGT Fan 1E is placed in RUN

If SGT-AO-251 (SGT Train A Outlet Valve) remains closed, how will SGT train B respond over the next one (1) minute?

"B" Standby Gas Treatment fan 1F . . .

a. remains off. SGT-AO-252 (SGT Train B Outlet Valve) opens.

b. starts on low flow. SGT-AO-252 (SGT Train B Outlet Valve) opens.

c. remains off. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

d. starts on low flow. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

Answer 36

c. remains off. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

Low flow in a train will cause the standby fan to start if it is in STBY, the operating train flow is <800 scfm, and a group 6 isolation signal is present or sealed in. There is no group 6 isolation signal for the conditions presented. No conditions will develop on the operating train to cause it to trip within 1 minute.

REFERENCE: 2.2.73

Foils:

a. SGT-AO-252 (SGT Train B Outlet Valve) does not open.
b. SGT B remains off.
d. SGT B remains off.

Modified
Difficulty 2
Coginitive Level 2

Enabling Objectives

COR0022802001080A   Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

COR0022802001090B   Briefly describe the following concepts as they apply to the Standby Gas Treatment system: Air operated valves operations

Skills

261000.K4.01   Knowledge of STANDBY GAS TREATMENT SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Automatic system initiation (3.7/3.8)

Question 37   1494   (1 point(s))

The turbine building HVAC was operating normally with all fan control switches in AUTO. A sudden atmospheric disturbance caused turbine building pressure to go 1" positive. The HVAC dP controller returned pressure to normal in 25 seconds.

What is the status of the turbine building supply and exhaust fans?

All supply fans . . .

a. **AND** exhaust fans are tripped.

b. **AND** exhaust fans remain running.

c. are tripped, all exhaust fans remain running.

d. remain running, all exhaust fans are tripped.

Answer 37

b. **AND** exhaust fans remain running.

REFERENCE: STCOR001-08-01, page 24, section II.F.3, Proc. 2.2.49, step 2.3 & 2.9

Foils:

a. 45 second delay on trip for excessive dp
c. 45 second delay on trip for excessive dp
d. 45 second delay on trip for excessive dp

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

COR0010802001220F   Given plant conditions, determine if the following should occur: Turbine Building Supply Fan trip

Skills

288000.A3.01 Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: (CFR: 41.7 / 45.7) Isolation/initiation signals (3.8/3.8)

Question 38   17901   (1 point(s))

With the plant at full power, the following Reactor Building vent exhaust plenum radiation monitor readings exist:

- RMP-RM-452A: 14 mrem/hr
- RMP-RM-452B: 7 mrem/hr
- RMP-RM-452C: 11 mrem/hr
- RMP-RM-452D: 13 mrem/hr

NO group isolations or automatic initiations occur.

What actions are required (if any) and why?
(Note: Use *actual* setpoints in your evaluation.)

a. **NO** actions are required because **neither** ***DIVISION*** logic has actuated.

b. **NO** actions are required because **only** the ***DIVISION I*** logic has actuated.

c. Manually start **only** "A" SGT train because **only** the ***DIVISION I*** logic has actuated.

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

Answer 38

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

Justification: If RMP-RM-452A or C AND RMP-RM-452B or D exceed 10 mrem/hr, Reactor Building isolates, and both SGT systems start. Per 2.0.3 "Operators shall validate automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals"

REFERENCE: 2.1.22, 2.2.73; 1.3.1.2 (logic), 2.3_9-4-1; Set Points

Foils:

a,b,c Both Divisions should have actuated. The reactor building should have isolated and both SGT trains should have started.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0022802001080A Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

COR0022802001130A Given plant conditions, determine if any of the following should occur: SGT automatic initiation

COR0011802001100D Given a control manipulation, predict and explain the changes to the following Radiation Monitoring systems: Reactor Building Vent Exhaust Plenum radiation monitoring system

COR0011802001120C Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: PCIS Group 6 Isolation

COR0011802001120D Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Standby Gas Treatment Startup

COR0011802001120E Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Reactor Building Ventilation Isolation

Skills

290001.A2.03 Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal...: (CFR: 41.5 / 45.6) High area radiation (3.4/3.6

Question 39   1112   (1 point(s))

DG-1 is paralleled to 4160V Bus 1F for testing. The operator is in the process of adjusting load and voltage when the Governor Control switch sticks in the LOWER position.

If NO operator action is taken, what will be the Diesel Generator response to this condition?

DG output frequency will . . .

a. lower **AND** then the diesel will trip on overcurrent.

b. lower **AND** then the diesel will trip on reverse power.

c. remain constant **BUT** the diesel will trip on overcurrent.

d. remain constant **BUT** the diesel will trip on reverse power.

Answer 39

d. remain constant **BUT** the diesel will trip on reverse power.

EXPLANATION OF ANSWER: Due to the diesel being paralleled the frequency will remain constant but load will be removed causing a reverse power trip which will trip the diesel generator lockout.
a,b. Frequency will remain constant.   c. Lowering on the governor control will remove load not increase load.

REFERENCES: STCOR002-08-02, page 34, section III.B.1, IV.4.C, rev. 10.
PR 2.3.2.8, page 4, section C-1/A-4, rev. 20.
PR 2.2.20, page 13, 8, section 4.4, 2.4.3.6, rev. 45.

Foils:

a. frequency will not lower and diesel will not trip on overcurrent.
b. frequency will not lower.
c. diesel will not trip on overcurrent.

Bank
Difficulty 2
Coginitive Level 3

Enabling Objectives

COR0020802001090A Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Diesel Generator Trips (Normal)

COR0020802001090C Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Speed Droop Control

COR0020802001130A Predict the consequences of the following items on the Diesel Generator: Parallel Operation of Diesel Generator

Skills

264000.K3.02 Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: (CFR: 41.7 / 45.4) A.C. electrical distribution (3.9/4.0)

Question 40   16439   (1 point(s))

The plant is at 100% power near the end of cycle with all control rods fully withdrawn.
When performing an APRM functional test (1/2 scram), the scram inlet valve (CRD-AOV-126) for control rod 30-31 opens.

What is the effect (if any) on reactor power and the Scram Discharge Volume (SDV) over the next five (5) minutes and why?

a. The reactor will scram due to high Scram Discharge Volume level.

b. Reactor power and SDV level are unaffected because **NO** control rod motion occurs.

c. The control rod inserts causing reactor power to lower, but power operation continues as SDV level is unaffected.

d. The control rod inserts causing reactor power to lower, but power operation continues and SDV level rises but remains below the scram setpoint.

Answer 40

c. The control rod inserts causing reactor power to lower, but power operation continues as SDV level is unaffected.

Control rod inserts. SDV level is unaffected because the over piston area volume is evacuated to the reactor vessel past the CRDM seals.

REFERENCE:
2.4CRD direct troubleshooting per 2.2.8.
2.2.8; section 25, Attachment 2 - 1.2.4.2

Foils:

a. A reactor scram will not occur. The SDV level will not change.
b. The control rod will insert into the core. A single control rod inserting will reduce reactor power.
d. No leakage will occur into the SDV.

Bank
Difficulty 3

Cognitive Level 2

Enabling Objectives

COR0020402001110H Predict the consequences a malfunction of the following would have on the CRDH systems: Leaking scram valves.

Skills

201003.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: (CFR: 41.5 / 45.5) Reactor power (3.7/3.8)

Question 41   16414   (1 point(s))

The plant is operating at 80% power. The CAM alarms in the Off-Gas Building and the following alarms are received:

- B-3/A-3, CONDENSER AIR REMOVAL ISOLATION
- B-3/B-3, CONDENSER AIR REMOVAL HIGH TEMP
- B-3/B-4, CONDENSER AIR REMOVAL HIGH PRESSURE
- K-1/A-4, OFFGAS FILTER HIGH D/P

What event has occurred?

a. High Off-Gas activity.

b. Fire in the Off-Gas Building.

c. Explosion in the Off-Gas System.

d. Trip of the Off-Gas Dilution Flow Fans.

Answer 41

c. Explosion in the Off-Gas System.

Justification: These alarms are consistent with an explosion in the off gas system and the CAM alarm for the hi radiation in the off gas building.

REFERENCE: 2.4OG 5.0; 2.3_B-3/A-3/B-3/B-4 section 3.2

Foils:

a. This would cause the isolation but not the high temperature and pressure.
b. This would not cause the isolation or the high pressure or the CAM alarm.
d. This would not cause the isolation or high temperature or the CAM alarm.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0011602001100G Predict the consequences of the following items on the Off Gas system: Condenser Air Removal failure

SKL012411600A030D Given plant conditions, predict changes in the following Off-Gas system components/parameters: System flows.

SKL012411600A030E Given plant conditions, predict changes in the following Off-Gas system components/parameters: Process radia tion monitoring indications.

Skills

271000.A3.05 Ability to monitor automatic operations of the OFFGAS SYSTEM including: (CFR: 41.7 / 45.7) System indicating lights and alarms (2.9/2.9)

Question 42   17884   (1 point(s))

The station was in a startup at 57% power when main condenser vacuum rapidly degraded. When vacuum degraded to 22.5" Hg, the CRS directed a manual reactor scram be inserted and then the main turbine be tripped.

Why was the reactor scrammed ***before*** the main turbine is tripped under these conditions?

To prevent . . .

a. forcing an automatic protective action.

b. a rapid depressurization of a critical reactor due to bypass valves opening fully.

c. the need to close the MSIVs which would add heat to the primary containment.

d. reducing MCPR below the operating limit due to the reduction in feedwater heating.

Answer 42

a. forcing an automatic protective action.

Reference 2.4VAC, Operations strategy (2.0.3, 2.0.1.2)

Foils:

b. this is not a basis for the scram.
c. The MSIVs still may need to be closed.
d. MCPR is not the basis for the scram.

NEW
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT0320132K0K0100 Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

INT0320132J0J0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

295002.AK3.01 Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM: (CFR: 41.5 / 45.6) Reactor SCRAM: Plant-Specific (3.7/3.8)

Question 43   17888   (1 point(s))

The plant was operating at 100% power when 4KV bus 1F de-energized. The reactor is scrammed per the alarm card. After the immediate scram actions were complete, the Startup Transformer tripped. The following conditions exist:

- Groups 2, 3 and 6 isolations occurred
- HPCI and RCIC did not automatically start
- 87 control rods are between notch 12 and notch 36
- APRM downscales are in

What action(s) (if any) are required and why?

a. Enter EOP 1A, "RPV Control". Do **NOT** enter 5.3EMPWR "Emergency Power" or 2.4CRD "CRD Trouble." Applicable Abnormal/Emergency procedures are **NOT** entered if EOP entry **IS** required.

b. Enter 5.3EMPWR "Emergency Power", 2.4CRD "CRD Trouble" **AND** EOP 1A, "RPV Control." Do **NOT** execute any Abnormal procedure steps if they conflict with any steps in the EOPs.

c. Enter 5.3EMPWR "Emergency Power" and 2.4CRD "CRD Trouble". Do **NOT** enter EOP 1A, "RPV Control" as EOP entry is **NOT** required if the specific condition is adequately addressed by the applicable Abnormal/Emergency procedure.

d. Enter EOP 1A, "RPV Control", 5.3EMPWR "Emergency Power" and 2.4CRD "CRD Trouble". EOPs and applicable Abnormal/Emergency procedures must be entered and ALL Abnormal and EOP procedure steps must be completed.

Answer 43

b. Enter 5.3EMPWR "Emergency Power", 2.4CRD "CRD Trouble" **AND** EOP 1A, "RPV Control." Do **NOT** execute any Abnormal procedure steps if they conflict with any steps in the EOPs.

REFERENCE: 2.0.1.2 step 2.7. 0.1, 5.8 step 3.2.1, 3.2.6

Foils:

a. Applicable Abnormal/Emergency procedures must be entered.
c. Applicable EOPs must be entered
d. ALL the AOP and EOP steps do NOT have to be performed if they are in conflict the EOPS take precedent.

K/A 295037 (SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown) 2.4.8
New
Cognitive Level: 2
Difficulty: 2

Enabling Objectives

INT032010100E0300 Describe the heirarchy between the Emergency Operating Procedures, Abnormal Procedures, and Emergency Procedures, including which guidance takes precedence.

Skills

2.4.8 Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs. (CFR: 41.10 / 43.5 / 45.13)

Question 44 4243 (1 point(s))

A weekly surveillance test was completed at 1300 on Tuesday of last week.

In accordance with procedure 0.26 (Surveillance Program), what is the **LATEST** this test can be scheduled to be completed this week, without exceeding an LCO?

a. 1300 on Wednesday

b. 0100 on Thursday

c. 0700 on Thursday

d. 1300 on Thursday

Answer 44

c. 0700 on Thursday

A deviaiton of 25% of the weekly surveillance interval is allowed (42 hrs or 1.75 days)

Foils:

a. 24 hrs
b. 36 hrs
d. 48 hrs

REFERENCE: PR 0.26 Surveillanve Program Page 10 Section 8.1.5

Bank
Difficulty 2
Cognitive Level: 1

Enabling Objectives

INT032010100G010J Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test performance

INT032010100G010M Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Frequency

INT032010100G010L Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Precautions and Limitations

Skills

2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Question 45   5131   (1 point(s))

Which one of the following activities can have its concurrent/independent verification waived?

a. A local control switch must be placed in Auto in a contaminated area as part of a tagout restoration.

b. A 4160 VAC breaker already verified in test will have a test block and extention arm installed.

c. A 480 VAC breaker for non-safety equipment will be racked out for a tagout requiring 25 mrem exposure to the performer.

d. A valve must verified open, requiring 12 mrem exposure to the performer when performing a periodic system component checklist.

Answer 45

d. A valve must verified open, requiring 12 mrem exposure to the performer when performing a periodic system component checklist.

Component checklist item with a dose exposure in excess of 10 mrem.

Reference: 0.31

Foils:

a. Required by procedure.
b. Required by procedure.
c. Not an exception.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

SKL00803020010200   Describe the three forms of verification used to satisfy the requirements of the equipment status control program and when each form of verification is required or permissible.

INT032010100H010C   Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: Independent/Concurrent verification

INT032010100H010L Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: Breaker position verification

Skills

2.1.29 Knowledge of how to conduct and verify valve lineups. (CFR: 41.10 / 45.1 / 45.12)

Question 46 16431 (1 point(s))

Following seven (7) days of vacation, an operator works twelve (12) hours on the first day back on shift and then works an additional four (4) hours of overtime.

Which one of the following describes the MAXIMUM number of hours this operator can work the next day WITHOUT exceeding the CNS working hour limitations?

a. 4 hours.

b. 8 hours.

c. 12 hours.

d. 16 hours.

Answer 46

b. 8 hours.

The restrictions are 16 hours in any 24-hour period and 24 hours in any 48-hour period. Since the operator worked 16 hours on the first day, then without any extension the operator is restricted to 8 hours on the second day of work.

REFERENCE: 0.12, 4.2.2

Foils:

a. Can work 8 hours.
c. Can only work 8 hours without an approved extension.
d. Can only work 8 hours without an approved extension.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT032010100F0100 State the working hours limitations and approval requirements associated with Administrative Procedure 0.12, Working Hours Limitations.

INT032010100F0200 Given the previous working hours/days history of an individual, determine if the individual is in compliance with the working hours limitations set forth in Administrative Procedure 0.12, Working Hours Limitations.

Skills

2.1.1 Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

Question 47   5654   (1 point(s))

A LOCA has occurred and the following conditions exist:

- RPV level is -120" and stable on the wide range instruments.
- Attempts at level restoration are in progress, but currently there are no means of injection available.
- Annunciator 9-3-1/A-1 ADS TIMERS ACTUATED is energized.
- RPV pressure is 490 psig and lowering.

What is the basis for inhibiting ADS at this time?

a. To prevent large amounts of uncontrolled injection into the RPV.

b. The ADS logic should not have actuated under current plant conditions.

c. The operator has more information as to when the RPV should be depressurized.

d. To allow the crew to rapidly depressurize the RPV to the main condenser and prevent adding significant heat to the suppression pool.

Answer 47

c. The operator has more information as to when the RPV should be depressurized.

REFERENCE: INT0080609 EOP basis, PR 2.3.2.22

Foils:

a. This is a basis for terminating and preventing injection during an ATWS.
b. These RPV pressure conditions would require ADS.
d. The existance of adequate core cooling at the time of ADS initiation is not a basis for inhibiting ADS.

Modified
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00806090011200 Given an EOP flowchart 1A, RPV CONTROL step, state the reason for the actions contained in the step.

Skills

2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 45.13)

Question 48   5349   (1 point(s))

A Loss of Coolant Accident has occurred with the following conditions:

- Reactor pressure	470 psig (lowering slowly)
- Indicated Wide Range Reactor water level	-120" (steady)
- Drywell pressure	5.5 psig (rising slowly)
- Drywell temperature	350° F (all points) (steady)

What is the status of Wide Range Reactor Level Instrumentation?

Wide Range Reactor Level Instrumentation is . . .

a. accurate **AND** can be used for trending.
Actual Reactor level is -120".

b. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>higher than</u> -120".

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>lower than</u> -120".

d. **NOT** accurate **AND CANNOT** be used for trending.

Answer 48

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is <u>lower than</u> -120".

EXPLANATION OF ANSWER: c. is correct. Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes. The elevated Drywell temperatures and increased density of RPV water combine to cause indicated level to be erroneously high and actual Reactor level will be lower than -120".

REFERENCES: EOP Graphs 1 & 15, CAUTION 1
INT0080618 INT0080605

Foils:

a. Indicated level is erroneously high and actual Reactor level will be lower than 120".
b. Actual Reactor level will be lower than -120".
d. Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes.

Provide to Candidate: EOP Graphs 15A, B, C, D & E

Bank
Difficulty 4
Cognitive Level 3

Enabling Objectives

INT00806050011000 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

INT00806180010200 For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

INT00806050011200 Given plant conditions, assess if RPV water level can be determined or not.

Skills

295028.EK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: CFR: 41.8 to 41.10) Reactor water level measurement (3.5/3.7)

Question 49   5268   (1 point(s))

Given the following conditions:

- A small steam leak has occurred in the Drywell.
- Drywell temperature is 165° F and rising.
- Drywell pressure is 2.6 psig and rising.
- EOP-3A, Primary Containment Control has been entered.

What action is required to reduce Drywell temperature for these conditions?

a. Vent the Drywell.

b. Initiate Drywell sprays.

c. Emergency Depressurize the RPV.

d. Operate all available Drywell Cooling.

Answer 49

d. Operate all available Drywell Cooling.

EXPLANATION OF ANSWER: This is the first action specified by the Drywell Temperature leg of EOP-3A.

REFERENCE: EOP-3A

Foils:

a. With a LOCA inside containment, venting the Drywell is not allowed.
b. Drywell Sprays are not permitted by the Drywell Spray Initiation Limit Graph.
c. Emergency Depressurization is not yet required.

Bank
Difficulty 2
Cognitive Level 2

Provide to Candidate: EOP 3A with entry conditions and CAUTIONS Removed.

Enabling Objectives

INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

INT00806130101000 Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Skills

295012.AK2.02 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8) Drywell cooling. (3.6/3.7)

Question 50 17856 (1 point(s))

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- RCIC cannot be isolated
- RCIC temperatures in the NE Quad are 214°F and rising
- RCIC Room radiation levels are 300 mr/hr and rising
- A reactor scram is inserted and all control rods fully insert

A high reactor building ventilation exhaust radiation level had isolated reactor building ventilation, but the condition has cleared and annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD reset.

What is the required EOP response to this annunciator clearing and the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Ensure Reactor Bldg. HVAC remains isolated because EOP 1A requires a group 6 isolation.

d. Ensure Reactor Bldg. HVAC remains isolated until RP ensures normal radiation levels to minimize the spread of contamination.

Answer 50

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

REFERENCE: EOP 5A, PSTG/SATG, 2.3_9-4-1

Foils:

a. This is <u>not</u> a bases for restarting Reactor Bldg. HVAC and Rx Bldg ventilation releases are NOT considered elevated.
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

Bank
Difficulty 3
Cognitive Level 2

Provide to Candidate: EOP 5A with entry conditions and Cautions removed.

Enabling Objectives

INT00806170010600 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

INT00806170010700 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Skills

295034.EK2.04 Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following: (CFR: 41.7 / 45.8) Secondary containment ventilation. (3.9/3.9)

Question 51   17157   (1 point(s))

The plant is operating at 100% power with "B" Narrow Range level indicator selected for level control. The following conditions occur:

- "A" and "C" Narrow range are 5" and lowering
- "B" Narrow range is 40" and rising
- Steam Nozzle is downscale

30 seconds later,

- "A" & "C" Narrow range are downscale
- "B" Narrow range is upscale
- "A" wide range is -10"
- "B" wide range is upscale
- ALL -42" and -113" alarms are clear
- Reactor Pressure 995 psi
- Drywell Temperature 125 °F

At the 30 second data time, can RPV water level be determined, and if so, what is actual RPV water level?

RPV water level _________ be determined. Actual RPV water level is ____________.

a.   CANNOT, unknown

b.   CAN, above +60"

c.   CAN, -10"

d.   CAN, below -42" and above -113"

Answer 51

c.   CAN, -10"

The conditions provided are indicative of a "B" side reference leg failure. There are multiple indications that RPV water level is above TAF.

REFERENCE: INT0080605, Attachment 2

Foils:

a. RPV water level CAN be determined
b. RPV water level is -10"
d. RPV water level is -10"

Bank
Difficulty 4
Cognitive Level 3

Enabling Objectives

INT00806090011300 Given plant conditions, assess if RPV water level can be determined or not.

INT00806050011200 Given plant conditions, assess if RPV water level can be determined or not.

Skills

295009.AA2.01 Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Reactor water level. (4.2/4.2)

Question 52   5920   (1 point(s))

Which of the following can be used to verify that the reactor will remain shut down under all conditions without boron ***without*** consulting the Reactor Engineer?

a. All control rods are inserted to or beyond notch 04.

b. One (1) control rod is at position 48, all other control rods are full in.

c. Two (2) control rods are at position 12, all other control rods are full in.

d. The withdrawn control rods are separated by at least two control rod cells in all directions.

Answer 52

b. One (1) control rod is at position 48, all other control rods are full in.

Reference: INT0080605

Foils:

a. Must be inserted to or beyond notch 02.
c. Shutdown margin may not be met. Not a condition specified to satisfy requirements.
d. Shutdown margin may not be met. Not a condition specified to satisfy requirements.

Bank
Difficulty: 2
Cognitive level: 1

Enabling Objectives

INT00806050010400   State the criteria used to determine that the reactor will remain shutdown under all conditions without boron injection.

Skills

295015.AK1.01 Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10) Shutdown margin. (3.6*/3.9*)

Question 53   16495   (1 point(s))

During a LOCA concurrent with a Loss of Offsite power, the following conditions exist:

- Drywell Pressure is 60 psig and slowly rising
- Torus Pressure is 58 psig and slowly rising
- PC water Level is 25 feet and stable

Which one of the following methods is required to ***initially*** vent the primary containment for these conditions?

a. Vent the Torus using the hard pipe vent.

b. Vent the Drywell using the 24" ductwork.

c. Vent the Torus using the 1 and 2 inch bypass valves through SGT.

d. Vent the Drywell using the 1 and 2 inch bypass valves through SGT.

Answer 53

c. Vent the Torus using the 1 and 2 inch bypass valves through SGT.

Justification: Per EOP-3A, Primary Containment Control, if Suppression Pool level is < 28.5', then vent the torus using the Torus Vent path and SGT.

Reference: EOP-3A, 5.8.18

Foils:

a. Hard pipe venting is NOT required.
b. The torus should be vented first, use of the 24" ductwork is NOT required.
d. The torus should be vented first.

Bank
Difficulty: 3
Cognitive level: 2

Enabling Objectives

INT00806130010500 Explain why the torus vent path is preferred over the drywell vent path.

INT00806130101000 Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

INT0080613001040A State the basis for primary containment control actions as they apply to the following: Specific setpoints

INT0080613001040B State the basis for primary containment control actions as they apply to the following: Primary Containment Control Systems

INT00806130011200 Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Skills

295010.AA1.05 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6) Drywell/suppression vent and purge (3.1/3.4)

Question 54   16481   (1 point(s))

Given the following conditions:

- The plant has experienced a LOCA with a loss of ALL injection.
- All Control Rods have inserted.
- RPV pressure is being controlled with SRVs.
- RPV water level has lowered to -35 inches (Corrected Fuel Zone).

What is the status of core cooling?

Adequate core cooling exists . . .

a. at this RPV water level.

b. only when the SRVs are closed.

c. only if RPV water level is raised 10 inches.

d. only if injection is established at this water level.

Answer 54

a. at this RPV water level.

At levels above -40 inches with no injection there is sufficient steam flow to provide adequate core cooling.

REFERENCE: EPGS/SATGS, INT008-06-09

Foils:

b. If SRVs close, adequate core cooling is NOT assured with RPV water level below TAF.
c. RPV level does not have to be raised to -25".
d. Just establishing injection will not assure adequate core cooling, level must also be restored to > -25".

Bank
Cognitive Level: 3

Difficulty: 3

Provide to Candidate: EOP 1A with entry conditions and Cautions removed.

Enabling Objectives

INT00806070010800 Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, state the reasons for the actions contained in the steps.

INT00806090010100 Describe the three mechanisms specified in the EOPs to assure adequate core cooling including the RPV water level band required and which is the preferred method.

Skills

295031.EA2.04 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Adequate core cooling (4.6*/4.8*)

Question 55   17900   (1 point(s))

Why is entry into EOPs required if the Reactor Building dP cannot be maintained negative?

This reactor building (RB) dP is an indication that . . .

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

b. the continued operability of equipment needed to carry out EOP actions may be compromised.

c. radioactivity is being released to the environment when the ventilation system should have automatically isolated.

d. an indication that water from a primary system (or from a primary to secondary system leak) may be discharging into the secondary containment.

Answer 55

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

REFERENCE: **INT0080617**

FOILS:
b. This is the basis for the high temperature entry.
c. This is the basis for the high Rx bldg exhaust radiation level.
d. This is the basis for the entry on radiation above Max Normal Operating Level.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00806170010100   List the entry conditions to Flowchart 5A (including the radioactivity release path) and briefly explain each.

Skills

2.3.11 Ability to control radiation releases. (CFR: 45.9 / 45.10)

Question 56   4139   (1 point(s))

What is the basis for stopping and preventing HPCI when suppression pool level lowers to 11 feet, irrespective of adequate core cooling?

a. Prevent HPCI pump damage from low NPSH.

b. Prevent uncovering the downcomers in the torus.

c. Prevent containment pressurization from HPCI exhaust.

d. Prevent loss of the ultimate heat sink by removing too much water.

Answer 56

c. Prevent containment pressurization from HPCI exhaust.

REFERENCE: EOP-3A, INT0080613

Foils:

a. HPCI protected for NPSH with low suction pressure trip.
b. This is the HCLL curve flat area at 9.6'.
d. Other level limits protect this 9.6' for downcomers, 6' for SRVs and HCTL and HCLL.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT00806130010900   Explain why HPCI but not RCIC must be secured at a primary containment water level of 11 feet.

Skills

295030.EK3.02 Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) HPCI operation: Plant-Specific. (3.5/3.7)

Question 57 17896 (1 point(s))

Why is the RPV Emergency Depressurized if Pressure Suppression Pressure (Graph 10) is exceeded?

a. Failure of primary containment may occur if a primary system rupture develops.

b. Failure of primary containment may occur if drywell sprays are initiated.

c. Failure of SRV Tailpipes may occur due to steam bypassing the suppression pool.

d. Failure of SRV Tailpipes may occur due to inadequate differential pressure across the balancing disc.

Answer 57

a. Failure of primary containment may occur if a primary system rupture develops.

REFERENCE: INT0080618, PSTG

Foils:

b. PSP is not based on drywell spray restricitons (DWSIL is).
c. SRVTPLL is not based on steam bypassing the suppression pool.
d. SRVTPLL is not based on dp on balancing disc (PCPL is).

Bank
Difficulty: 3
Cognitive level: 1

Enabling Objectives

INT00806130011200 Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

INT0080613001040C State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A

INT00806180010200 For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Skills

295024.EK3.04 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.5 / 45.6) ?Emergency depressurization (3.7/4.1)

Question 58 16482 (1 point(s))

Following a control room evacuation due to toxic gas in the control room, operations personnel are at the stations required by the associated procedure.

Who verifies RPV level and RPV pressure?

a. The Control Room Supervisor.

b. The Reactor Building Station Operator.

c. The Control Building/Critical Switchgear Operator.

d. The Turbine Building Switchgear A and B Operators.

Answer 58

a. The Control Room Supervisor.

Justification: Verified at the ASD Panel, which is manned by the SS and CRS.

REFERENCE: 5.1ASD

Foils:

b. Operates SW Valve
c. Checks Critical Buses and isolates RCIC
d. Determines status of Non-critical 4 KV Buses

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

OTH01592020010600 Given a standard set of references available in the Control Room, be able to perform the steps necessary to shutdown the reactor from outside the control room as the ASD room operator.

Skills

295016.AA1.06 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.7 / 45.6) Reactor water level. (4.0/4.1)

Question 59   17858   (1 point(s))

The plant is at 60% power for a rod sequence change.

The RO has just withdrawn control rod 22-19 to notch 24. After the rod control movement switch is released and the SETTLE light turns off, the RO observes control rods 22-19 slowly moving inward. No other control rods are moving.

The following CRD parameters exist:

- CRD Drive Water D/P   350 psid (pegged high)
- CRD Cooling Water flow   60 gpm (pegged high)

What action is required?

a. Scram the reactor.

b. Using EMERGENCY IN, fully insert rod 22-19.

c. Momentarily place the rod movement control switch to WITHDRAW.

d. Using CRD-MO-20, DRIVE PRESSURE CONT VALVE, reduce drive water differential pressure.

Answer 59

b. Using EMERGENCY IN, fully insert rod 22-19.

REFERENCE: 2.4CRD

Foils:

a. The reactor would be scramed if two or more rods were drifting in, not for a single rod.
c. This may temporarily stop the rod movement, but the rod must be driven in.
d. Drive water D/P is high but the action required is to insert the control rod, then diagnose the cause of the rod drift.

Modified
Difficulty 3
Cognitive Level 2

Enabling Objectives

INT0320122I0I0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

INT0320122J0J0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

201001.A2.12 Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those ...: (CFR: 41.5/45.6) High cooling (2.8/2.9)

Question 60   16490   (1 point(s))

The reactor has been shutdown for 18 hours and is currently in Cold Shutdown (MODE 4). A cooldown is in progress with reactor coolant temperature at 162°F. RHR Loop "A" is in Shutdown Cooling with both reactor recirculation pumps tripped. Subsequently, a Group 2 isolation signal occurs and RHR CANNOT be restarted.

Where is RPV water level required to be maintained for the current conditions and why?

a. Above +48 inches on the narrow range RPV level instruments to promote natural circulation.

b. At 0.0 inches on the wide range RPV level instruments to support alternate heat removal using RWCU.

c. Flooded (solid) on the shutdown range RPV level instruments to support alternate heat removal using the SRVs.

d. Between +27.5 inches and +42.5 inches on the narrow range RPV level instruments to minimize thermal stratification in the reactor pressure vessel.

Answer 60

a. Above +48 inches on the narrow range RPV level instruments to promote natural circulation.

REFERENCE:
2.4SDC; 4.9, and attachment 2 (Step 4.4 to step 4.9 to Attachment 2)
2.1.4 precaution 2.9

Foils:

b. Water level is not high enough to support this method of heat removal
c. Not an approved method of heat removal under these conditions.
d. Circulation is needed to minimize thermal stratification.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT032010400B0400   Discuss cautions and notes associated with Procedure 2.1.4, Normal Shutdown.

INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

295021.AK3.05 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5 / 45.6) Establishing alternate heat removal flow paths (3.6/3.8)

Question 61   16468   (1 point(s))

The unit is operating at 100% power when the following alarms are energized:

- M-1/A-1, REC SYSTEM LOW PRESSURE
- M-1/A-3, REC SURGE TANK LOW LEVEL

Fifty (50) seconds after receipt of the alarms, the appropriate immediate operator actions have been completed. REC pressure is 56 psig and lowering.

What action(s) is/are now required?

a. Trip ALL Drywell FCUs.

b. Isolate REC to the Augmented Radwaste System.

c. Manually scram the reactor and shutdown ALL REC Pumps.

d. Perform rapid power reduction per 2.1.10 to 35 Mlbm/hr core flow.

Answer 61

c. Manually scram the reactor and shutdown ALL REC Pumps.

Justification: These alarms indicate a break in the REC piping that will exceed makeup capacity and result in a loss of ability to cool the recirc pumps and reactor auxiliary equipment. The system must be shutdown, the reactor scrammed, and then limited cooling may be accomplished with one pump.

REFERENCE: 5.2REC 4.1, 4.7 Attachment 2, Step 1.1

Foils:

a. There is no requirement to secure the FCUs.
b. This valve automatically isolates when REC pressure falls to 60.5 psig for 40 seconds which was satisfied based on the alarm condition present for 50 seconds.
d. This would be required by 2.4PC if not for the reactor scram that is required first.

K/A 400000 Component Cooling Water

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT0320126O0O0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question 62   13338   (1 point(s))

With the plant operating at 100% power, a loss of off site power occurs. Both diesel generators fail to start and CANNOT be started. HPCI and RCIC recover RPV water level to +35" (NR), when flow is stopped on both systems. RPV water level is lowering at an average of 1 inch per minute due to SRV operation.

What procedural restriction (if any) applies to the continued use of HPCI in response to this event until on-site or off-site electrical power can be restored?

a. HPCI suction must be shifted to the suppression pool.

b. HPCI must be placed in RPV Pressure Control Mode.

c. HPCI can be operated until the division 2 battery is exhausted.

d. HPCI must be secured after one cycle of operation and must remain off.

Answer 62

d. HPCI must be secured after one cycle of operation and must remain off.

This procedure assumes that RPV water level and pressure is initially controlled by HPCI, as directed by the EOPs. CNS has committed to secure HPCI after one cycle of operation, even if EOPs ***allow*** HPCI use, in order to extend station battery life during station blackout. HPCI would still be operated if ***required*** by EOPs, but the conditions provided are within RCIC capability.

REFERENCES: 5.3SBO; 1.4 and NOTE, and 5.3

Foils:
a. There is no requirement to shift to the SP
b. HPCI must be secured to save DC power.
c. HPCI must be secured to save DC power.

Modified
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT0060119001150D Describe each of the following special events evaluated in the CNS USAR that could challenge the integrity of the radioactive material barriers: Station Blackout

INT0320131W0W0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

INT00601190012200 Given a specific USAR analyzed Special Event, describe the initial plant condition assumed in the analysis.

INT00601190012500 Given a specific analyzed Anticipated Operational Transient or Special Event, state the appropriate safety actions necessary to prevent exceeding their safety design bases.

Skills

295003.AK1.06 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10) Station blackout: Plant-Specific. (3.8/4.0*)

Question 63   3175   (1 point(s))

Given the following conditions:

- The Plant is operating at 100% power.
- A fire in the Cable Spreading Room has been reported **AND** the Fire Brigade is on the scene.
- The actions of 5.4FIRE, "General Fire Procedure" **AND** 5.4POST-FIRE, "Post-Fire Operational Information" are being performed.
- The Control Room Supervisor enters 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room" as directed by procedure.

Based on these conditions, which choice below describes a condition that will dictate a Turbine Trip be initiated per 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"?

A Turbine Trip should be performed . . .

a. prior to leaving the Control Room, **IF** possible.

b. **ONLY** for a confirmed fire in the Control Room.

c. **ONLY IF** the fire in the Cable Spreading Room is confirmed.

d. **ONLY IF** unexpected DEH/Turbine related actuations occur.

Answer 63

a. prior to leaving the Control Room, **IF** possible.

REFERENCE: PR 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"

Foils:

b. Required prior to leaving if possible, irrespective of fire location.
c. Not a reason to trip the turbine.
d. Not a reason to trip the turbine.

Bank

Difficulty 2
Cognitive Level 1

Enabling Objectives

INT0320134G0G0100 Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

Skills

2.4.27 Knowledge of fire in the plant procedure. (CFR: 41.10 / 43.5 / 45.13)

Question 64   17850   (1 point(s))

A plant shutdown is in progress at 88% power when the following indications are received:

- Steam line flow on the "D" steam line indicates zero (0) flow.
- Annunciator 9-5-2/F-2, REACTOR HIGH PRESSURE alarms.
- Reactor pressure continues to rise.

What action is required?

a. Lower the DEH pressure setpoint.

b. Place Governor valves in MANUAL.

c. Manually scram the reactor at 1030 psig.

d. If only one (1) MSIV is closed, re-open the valve.

Answer 64

c. Manually scram the reactor at 1030 psig.

Manually scram the reactor at 1030 psig per 2.4DEH and 2.3_9-5-2.

Reference 2.4DEH and 2.3_9-5-2.

Foils:

a. This would have no effect and is not in accordance with any procedure.
b. This would not be appropriate and would not correct the problem.
d. An MSIV should not be re-opened without engineering concurrence.

K/A 295025 (High Reactor Pressure), 2.4.11 (3.4)

Bank
Difficulty: 4
Cognitive level: 3

Enabling Objectives

INT0320125L0L0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Question 65   10634   (1 point(s))

The plant was operating at 100% power when the following annunciators alarmed:

- A-4/B-5, SERVICE AIR LOW PRESSURE
- A-4/B-4 SERVICE AIR ISOLATION PCV-609
- A-4/A-4, AIR RECEIVER A OR B LOW PRESSURE alarmed

The crew noted INSTRUMENT AIR PRESSURE (IA-PI-606), indicated 75 psig and closed IA-MO-80, NON CRIT INSTRUMENT AIR ISOLATION. Immediately following the closure of IA-MO-80 control room indications are:

- SERVICE AIR PRESSURE (SA-PI-611), indicates 0 psig.
- INSTRUMENT AIR PRESSURE (IA-PI-606), indicates 90 psig and increasing.

Which operator actions(s) is/are required next?

a. Re-open SA-PCV-609 to restore Service Air.

b. Re-open IA-MO-80 to restore non-critical Instrument Air.

c. Scram the reactor and transfer level control to HPCI/RCIC.

d. Manually open "A" Air Compressor TEC Return Isolation, TEC-AOV-21AV.

Answer 65

c. Scram the reactor and transfer level control to HPCI/RCIC.

Even though the closure of IA-MO-80 restored the instrument air pressure the loss of non-critical instrument air supply causes unsafe operation due to the loss of feedwater heating. The reactor should be scrammed as a scram required setpoint has been exceeded, even though air pressure is now above that requiring a scram.

REFERENCE: 5.2AIR; 2.1.5, Attachment 2   2.0.1.2 Step 7.1.4
SER 4-00

Foils:

a. is incorrect. Even though this action could be performed to safely restore service air, a scram is required.

b. is incorrect. Even though this action could be performed to safely restore instrument air, a scram is required.

d. is incorrect. This is a valve that fails closed on a loss of air, but there is no guidance to manually open the valve.

K/A 295019 (Part. Or Comp. Loss of Inst. Air) 2.4.31
Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

INT0320136N0N0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question 66   7755   (1 point(s))

Which one of the following is the normal, 100% power, lineup of the Containment H_2 and O_2 Monitoring system per 2.2.60.1 and what indications are available?

a. One Division in operation provides **only** O_2 measurement ***capability*** consisting of PMIS displays, O_2 recorders, an O_2 digital indicator.

b. One Division in operation provides H_2 **AND** O_2 measurement ***capability*** consisting of PMIS displays, H_2 **AND** O_2 recorders, an O_2 digital indicator and H_2 annunciation.

c. Two divisions in operation. Both provide H_2 **AND** O_2 monitoring consisting of H_2 **AND** O_2 recorders, a digital O_2 indicator and H_2 annunciation. **NO** input is provided to PMIS.

d. Two divisions in operation. One division provides H_2 monitoring, the other provides O_2 monitoring. An O_2 recorder, digital O_2 indicator and H_2 annunciation are provided. Both divisions provide input into PMIS displays.

Answer 66

b. One Division in operation provides H_2 **AND** O_2 measurement ***capability*** consisting of PMIS displays, H_2 **AND** O_2 recorders, an O_2 digital indicator and H_2 annunciation.

Foils:

a. H2 monitoring is available and annunciated.
c. Only one division in service, PMIS input is provided.
d. Only one division in service monitoring both parameters.

Reference: 2.2.60.1 Discussion Section.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0020302001140D Briefly describe the following concepts as they apply to the Primary containment: Hydrogen/oxygen concentration measurement

Skills

500000.EA1.02 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6) Primary containment oxygen instrumentation (3.3/3.2)

500000.EA1.01 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6) Primary containment hydrogen instrumentation (3.4/3.3)

Question 67 17854 (1 point(s))

The plant is operating at rated power when one (1) inboard MSIV fast closes. What is the expected plant response and why?

Reactor power and pressure will rise due to . . .

a. decreased control rod worths. The reactor will scram.

b. feedback from the void coefficient of reactivity. The reactor will scram.

c. decreased control rod worths. Reactor power will stabilize above 100%, but below the scram setpoint.

d. feedback from the void coefficient of reactivity. Reactor power will stabilize above 100%, but below the scram setpoint.

Answer 67

b. feedback from the void coefficient of reactivity. The reactor will scram.

REFERENCE: Reactor theory, USAR IV-4.

Foils:

a. Reactor pressure will rise because the steam lines have isolated.
c. Reactor power will lower because a scram occurs on an MSIV isolation on low pressure in RUN (<835 psig).
d. Reactor power will lower because a scram occurs on an MSIV isolation on low pressure in RUN (<835 psig)
Reactor pressure will rise because the steam lines have isolated.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT00601190010100 Given a set of initial operating conditions, select those conditions that would tend to make the consequences of an analyzed Anticipated Operational Transient more severe.

COR0021402001060B Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor pressure

INT0060119001050A For each of the following Anticipated Operational Transient categories listed, select an example of each transient, as stated in the CNS USAR: Nuclear system pressure increase

COR0021402001060D Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor power

Skills

295020.AK3.04 Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.5 / 45.6) Reactor pressure response. (4.1/4.1)

Question 68   13752   (1 point(s))

During a LOCA, the CRS requests a round of primary containment parameters. The RO reports the following:

- DW pressure: 4.5 psig and rising
- DW temperature: 160°F and rising
- Torus water level: 0 inches
- Torus water temp: 85°F

Does the report made by the RO meet the requirement(s) for the requested information and why or why not?

a. Meets. **ONLY** values are required.

b. Meets. **ONLY** changing parameters require trends or rate of trend.

c. **NOT** meet. Trend and rate of trend is required for the identified parameters.

d. **NOT** meet. Trends are required for ALL of the identified parameters. Rate of trend is not required.

Answer 68

c. **NOT** meet. Trend and rate of trend is required for the identified parameters.

During execution of EOPs, the CRS is expected to periodically direct one CRO to provide a round of primary containment parameters. Upon request, the parameter, trend, and rate of trend are to be included with no exceptions. Per 2.0.3, step 7.3.7, only the parameter and trend are normally required but since a "round of parameters" was requested, the rate of trend is also required.

REFERENCE: 2.0.3, step 8.6

Foils:

a. Does not meet standards. Trends and rate of trends are required.
b. Does not meet standards. Trends and rate of trends are required for parameters that are stable.

d. Rate of trend is required.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

OTH0151003001010D From memory define the following terms in accordance with in procedure 2.0.3, Conduct of Operations, and Operations Instruction #7: Briefs and updates

Skills

2.4.12 Knowledge of general operating crew responsibilities during emergency operations. (CFR: 41.10 / 45.12)

Question 69   17873   (1 point(s))

Which one of the following is the FEDERAL LIMIT for annual whole body dose to a control room operator in a calendar year?

a. 4 rem/calendar year

b. 5 rem/calendar year

c. 15 rem/calendar year

d. 50 rem/calendar year

Answer 69

b. 5 rem/calendar year

The federal limit is 5 rem/year.

Reference: 10CFR20, 9.ALARA.1

Foils:

a. 4 Rem is a CNS administrative limit.
c. 15 Rem is the federal limit for dose to the lens of the eye.
d. 50 Rem is the federal limit for dose to the extremities.

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

INT03201160000100 State the CNS administrative and federal exposure limits

Skills

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10)

Question 70   6047   (1 point(s))

A Site Area Emergency has been declared at CNS. Which Emergency Response Facility provides the location for the overall response to an emergency, once all the Emergency Response Facilities are activated?

a. Control Room

b. Technical Support Center.

c. Operations Support Center.

d. Emergency Operations Facility.

Answer 70

d. Emergency Operations Facility

Reference: 5.7.9

Foils:

a. The Control room manages the plant the EOF provides the overall response.
b. The TSS provides technical support to the EOF
c. The OSC provides support to the EOF.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

GEN0030401B0B040B Emergency Response Facilities (ERFs): b) Describe the functions of each of the following ERFs: 1) Control Room (CR) 2) Technical Support Center (TSC) 3) Operation Support Center (OSC) 4) Emergency Operations Facility (EOF) 5) Joint Information Center (JIC)

Skills

2.4.29 Knowledge of the emergency plan. (CFR: 43.5 / 45.11)

Question 71 3144 (1 point(s))

Given the following conditions:

- The plant is operating at 100% power.
- High Pressure Coolant Injection (HPCI) is in service for the quarterly full-flow test.
- Initial Suppression Pool Temperature is 92°F.

As HPCI continues to operate, what Technical Specifications actions are required?

When Suppression Pool Temperature *exceeds* …

a. 100°F, reduce thermal power to ≤ 1% Rated Thermal Power

b. 100°F, verify Suppression Pool temperature is less than 120°F once every 30 minutes.

c. 105°F, remove HPCI from service.

d. 105°F, place the Reactor Mode Switch in SHUTDOWN.

Answer 71

c. 105°F, remove HPCI from service.

REFERENCE: T.S 3.6.2.1

Foils:

a. Not required until completion times not satisfied.
b. Not required until 110°F
d. Not required until 110°F

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT00705070010400 From memory, in MODES 1,2, and 3, state the actions required in less than one hour if suppression pool average temperature exceeds 105 degrees F and THERMAL POWER IS > 1% RAP and performing testing that adds heat to the suppression pool (LCO 3.6.2.1)

Skills

295013.AA2.01 Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.8/4.0)

Question 72 17855 (1 point(s))

With the plant operating at full power, what is the basis for the Technical Specification requirement to place the Reactor Mode Switch in SHUTDOWN on high suppression pool temperature?

a. Ensure that low pressure ECCS pump NPSH requirements are met during a design bases LOCA.

b. Ensure that low pressure ECCS pump Vortex limits re not exceeded during a design bases LOCA.

c. Ensures that unstable condensation (chugging) does not occur in the suppression pool during a design bases LOCA.

d. Ensures the reactor is shutdown to prevent the suppression pool from being heated beyond design limits by the steam generated during a design bases LOCA.

Answer 72

d. Ensures the reactor is shutdown to prevent the suppression pool from being heated beyond design limits by the steam generated during a design bases LOCA.

REFERENCE: T.S. 3.6.2.1 Bases

Foils:

a. NPSH is an EOP related concept, not directly a Tech Spec limit bases.
b. Vortex limits is an EOP related concept, not a Tech Spec limit bases.
c. Chugging is an EOP related concept, not a Tech Spec limit bases.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00705070010500 From memory in MODES 1,2, and 3, state the actions required in less than one hour if suppression pool average temperature > 110 degrees F but is either less than or equal to 120 degrees F (LCO 3.6.2.1)

INT00705070010200 Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Skills

295026.EK3.05 Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.5 / 45.6) Reactor SCRAM. (3.9/4.1)

Question 73   17892   (1 point(s))

The plant is at 69% power with MAIN STEAM ISOLATION VALVE OPERABILITY TEST (IST) in progress. Inboard MSIV 80A will be tested. The operator depresses MSIV 80A TEST button and HOLDS it in the depressed state.

What will be the ***final*** valve position and Reactor Pressure response for the action above?

The MSIV . . .

a. partially closes. Reactor pressure rises above the scram setpoint.

b. partially closes. Reactor pressure rises and stabilizes below the scram setpoint.

c. fully closes. Reactor pressure rises above the scram setpoint.

d. fully closes. Reactor pressure rises slightly and stabilizes below the scram setpoint.

Answer 73

d. fully closes. Reactor pressure rises slightly and stabilizes below the scram setpoint.

REFERENCE: COR0021402, 6.MS.201

Foils:

a. The MSIV goes fully closed, pressure will rise slightly, not above scram setpoint.
b. The MSIV goes fully closed.
c. Pressure will rise slightly, not above scram setpoint.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

SKL012421400A030A   Given plant conditions, predict changes in the following Main Steam system components/parameters: System pressure

SKL012421400A030F Given plant conditions, predict changes in the following Main Steam system components/parameters: MSIVs

SKL012421400A030D Given plant conditions, predict changes in the following Main Steam system components/parameters: Reactor pressure

Skills

239001.A4.04 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System pressure (3.8/3.7)

Question 74 3963 (1 point(s))

The plant is at 75% power when a LOSS of the Normal Station Transformer occurs.

- EXCEPT for Bus 1B, ALL 4160 VAC buses transferred to the Startup Transformer successfully
- Due to a relay failure, breaker 1BS does not close automatically until two (2) seconds after the loss of the Normal transformer.
- Bus 1B is currently energized via 1BS

What is the state (running or stopped) of the Condensate (COND) and Condensate Booster (CB) pumps ***powered from Bus 1B*** five (5) seconds after Bus 1B is energized?

	COND PUMPS	CB PUMPS
a.	RUNNING	RUNNING
b.	RUNNING	STOPPED
c.	STOPPED	RUNNING
d.	STOPPED	STOPPED

Answer 74

a. RUNNING RUNNING

CP and CBP breakers do not have UV protection and will remain closed upon loss of power.

REFERENCE: 2.4MC-CF 2-2; 2.3_C-3/A-7 2.1.3.3, 2.1.3.4; 2.2.6 2.3
STCOR0020202 Condensate and Feed Page 20 Section II.B Rev 11; PR

Foils:

b.c.d. Both CB and CBP breakers remain closed and the pumps will be running after the bus is re-energized. The 1 second is much shorter than the low suction pressure trip setpoint for the CBP.

Bank
Difficulty 2

Cognitive Level 2

Enabling Objectives

COR0020202001080E Predict the consequences a malfunction of the following would have on the Condensate and Feedwater system: AC Power

Skills

259001.K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: (CFR: 41.7 / 45.7) A.C. electrical power (2.9/3.1)

Question 75   16412   (1 point(s))

The plant is at **75%** power. The Reactor Level Control system is maintaining RPV level at +35 inches (NR) in three (3) element control. The "C" main steam flow transmitter output goes to ZERO and remains at zero.

How will RPV water level change (lower or rise) and what will be the magnitude of the change?

RPV water level . . .

a. LOWERS and the reactor scrams on low level.

b. LOWERS and stabilizes at approximately +26 inches (NR).

c. RISES and stabilizes at approximately +44 inches (NR).

d. RISES and the reactor scrams when the main turbine trips.

Answer 75

b. LOWERS and stabilizes at approximately +26 inches (NR).

Level lowers until it stabilizes at 26 inches. A reactor scram will not occur. The magnitude of the level change is driven by the initial power level. 100% = 12" change, 75% = 9" change.

REFERENCE: 2.4RXLVL; 5.7

Foils:

a. The reactor will not scram.
c. Level lowers. It will rise if a feedwater transmitter failed low.
d. Level lowers. It will rise if a feedwater transmitter failed low.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0023202001090D   Given a specific RVLC system malfunction determine the effect on any of the following: RPV water level

COR0023202001060I   Predict the consequences of the following on the RVLC system: Loss of one or more main steam flow inputs

Skills

259002.A3.03   Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: (CFR: 41.7 / 45.7) Changes in main steam flow (3.2/3.2)

Question 76   5103   (1 point(s))

Given the following conditions:

- The reactor has been shut down for 12 days, with the reactor cavity flooded to 1001
- A complete loss of Reactor Equipment Cooling (REC) flow occurs
- It is unknown when REC System flow will be restored

Which method of cooling the Fuel Pool is available?

Fuel Pool cooling will be provided by . . .

a. the cross-tie with Circulating Water (CW) System.

b. the cross-tie with Turbine Equipment Cooling (TEC) System.

c. the cross-tie with the Residual Heat Removal (RHR) System.

d. the Service Water Cross-tie to the Fuel Pool Cooling Heat Exchanger.

Answer 76

c. the cross-tie with the Residual Heat Removal (RHR) System.

RHR cross-tie is available in the Shutdown Cooling Mode.

REFERENCES: COR001-06-02, page 21-23, section IV, V PR 2.4.8.6, page 1, section 4, PR 2.2.69.2, page 33, section 14

Foils:

a. Circ Water cross-ties with Service Water, but does not cross-tie with FPC.
b. REC provides backup cooling for some REC loads but not for FPC.
d. Service Water is used as an emergency makeup source but cannot be aligned to cool the FPC HX.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0010602001050A Describe the interrelationship between the FPC system and the following: Reactor Equipment Cooling

COR0010602001050B Describe the interrelationship between the FPC system and the following: Residual Heat Removal

COR0010602001080E Predict the consequences a malfunction of the following would have on the FPC system: REC

Skills

233000.K1.02 Knowledge of the physical connections and/or cause- effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Residual heat removal system: Plant-Specific (2.9/3.0)

Question 77   16409   (1 point(s))

NOTE: COR002-21-02, Figure 3 (RPS Trip System A) is provided for use with this question.

From rated conditions, a power excursion to 125% occurs. The following annunciators are received:

- 9-5-2/A-3, REACTOR SCRAM CHANNEL B
- 9-5-2/B-1, NEUTRON MONITORING TRIP

NO control rods moved. At the 9-5 vertical panel, you observe the following:

- White CRD Scram Solenoid Group lights for RPS Trip System "A" are ON
- White CRD Scram Solenoid Group lights for RPS Trip System "B" are OFF

NO operator actions have been taken in response to the conditions stated above.
If the 5A-K15A and the 5A-K15C relays will NOT change state, which one of the following operator actions will cause <u>ALL</u> control rods to fully insert?

a. Depressing the "A" manual scram pushbutton.

b. Placing the Reactor Mode Switch to SHUTDOWN.

c. Resetting RPS and then inserting a manual reactor scram.

d. Placing "A" and "C" RPS trip channel test switches to TRIP.

Answer 77

d. Placing "A" and "C" RPS trip channel test switches to TRIP.

This will directly remove power from the K14 relays which will de-energize the 117/118 solenoids at the HCU and scram the rods.

REFERENCE: RPS prints

RPS Trip System A Figure (COR002-21-02, Figure 3)

Foils:

a. K15A and K15C must both actuate to insert all control rods. This action will de-energize the K-15 relays which will not change state. If they don't drop out, the reactor will scram.
b. K15A and xK15C must both actuate to insert all control rods. This action will de-energize the K-15 relays which will not change state. If they don't drop out, the reactor will scram.
c. K15A and K15C must both actuate to insert all control rods. This action will de-energize the K-15 relays which will not change state. If they don't drop out, the reactor will scram.

Bank
Difficulty 4
Cognitive Level 3

Provide to Candidate: RPS Trip System A Figure (COR002-21-02, Figure 3)

Enabling Objectives

COR0022102001050A Briefly describe the following concepts as they apply to RPS: Logic arrangements

COR0022102001080F Given a specific RPS malfunction, determine the effect on any of the following: RPS logic channels

Skills

212000.A2.21 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control...: (CFR: 41.5 / 45.6) ?Failure of individual relays to repositition: Plant-Specific (3.6/3.9)

Question 78   1071   (1 point(s))

Given the following:

- A Reactor startup in progress.
- All IRMs are on range 9 and reading approximately 10 on the lower (Red) scale.
- The Reactor Mode switch is in START & HOT STBY.
- IRM A fails downscale.

What is the Rod Block status for the present conditions **AND** if the Reactor Mode Switch is placed in RUN?

a. **NO** Rod Block exists.
Placing the Reactor Mode switch in RUN **WILL** result in a Rod Block.

b. **NO** Rod Block Exists.
Placing the Reactor Mode switch in RUN will **NOT** result in a Rod Block.

c. A rod Block exists.
The Rod Block **WILL** clear after placing the Reactor Mode switch in RUN.

d. A Rod Block exists.
The Rod Block will **NOT** clear after placing the Reactor Mode switch in RUN.

Answer 78

c. A rod Block exists.
The Rod Block **WILL** clear after placing the Reactor Mode switch in RUN.

EXPLANATION OF ANSWER: An IRM downscale will generate a rod block. All IRM SCRAMs, Rod Blocks and Alarms are bypassed when the Mode Switch is in RUN, except Upscale or INOP with the companion APRM downscale.

Foils:
a,b. An IRM downscale will generate a rod block with the Mode Switch not in RUN.
d. All IRM SCRAMs, Rod Blocks and Alarms are bypassed when the mode switch is in RUN, except Upscale or IPOP with the companion APRM Downscale.

REFERENCES: STCOR002-12-02, 4.1.2

Bank

Difficulty 2
Cognitive Level 3

Enabling Objectives

COR0021202001060C Given a specific IRM malfunction, determine the effect on any of the following: Reactor manual control system

Skills

215003.K3.02 Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor manual control (3.6/3.6)

Question 79   17852   (1 point(s))

The station is in Mode 2 withdrawing control rods in an approach to criticality during a startup. The following equipment simultaneously trips:

- IRM "A", "C", "E" and "G"
- SRM "A" and "C"
- Off-Gas Radiation monitor "A"
- Reactor Building Vent Radiation monitors "A" and "C"

Control rods remain at their pre-transient position and NO group isolations have occurred.

Which one of the following conditions exists and what action (if any) is required?

a. A loss of RPSPP "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

b. A loss of RPSPP "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

c. A loss of 24 VDC "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

Answer 79

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

The loss of 24 vdc will cause all these instruments to become inoperative. No scram or group isolation will occur due to this single power loss.

REFERENCE: 2.2.26

Foils:

a. RPS power loss would not cause the loss of IRMs/SRMs. No ATWS or group isolation failure has occurred.
b. RPS power loss would not cause the loss of IRMs/SRMs.
c. No ATWS or group isolation failure has occurred.

K/A 295004 (295004 Partial or Complete Loss of DC Power), 2.4.48 ((3.5)

New
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0020702001080B Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)

COR0020702001080J Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor Protection system

COR0021202001070B Predict the consequences of a loss or malfunction of the following would have on the IRM system: 4/48 VDC

COR0020702001080K Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: IRMs

COR0020702001080L Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: SRMs

COR0020702001080R Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Radiation Monitoring systems

Skills

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. (CFR: 43.5 / 45.12)

Question 80 715 (1 point(s))

The plant is at power during an ATWS. An SRV closes after automatically lifting on pressure setpoint.

What effect does the SRV closing have on reactor power and why?

Power will . . .

a. rise due to reduction in core inlet subcooling.

b. rise due to collapse of voids in the core region.

c. lower due to reduction in core inlet subcooling.

d. lower due to collapse of voids in the core region.

Answer 80

b. rise due to collapse of voids in the core region.

Reference 2.4SRV

Foils:

a power rise is not due to inlet subcooling.
c. power will rise and is not due to inlet subcooling
d. power will rise

Modified
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0021602001040F Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Reactor power

Skills

295007.AK1.03 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Pressure effects on reactor power (3.8/3.9)

Question 81   17899   (1 point(s))

Given the following conditions:

- Reactor is in Hot Shutdown
- DG 1 is paralleled to 1F for surveillance testing
- The diesel governor fails to maximum fuel rack position

Which breaker(s) will trip (if any) as a result (direct or indirect) of the high DG 1 current?

a. **ONLY** 1AF

b. **ONLY** 1FA

c. **BOTH** 1AF and 1FA

d. **NEITHER** 1AF or 1FA

Answer 81

b. **ONLY** 1FA

EXPLANATION OF ANSWER: 1FA is tripped by the over current condition. 1AF will not trip because 1AF is in NORMAL AFTER CLOSE and Bus 1A is energized.

REFERENCES: STCOR001-01-02
PR 2.2.18, page 13, 15, section 4.6.2, 4.9.2.4
PR 2.2.20, page 16, section 4.9

Foils:

a. **only** 1FA trips on overcurrent
c. **only** 1FA trips on overcurrent
d. **only** 1FA trips on overcurrent

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0010102001090B Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Circuit breaker automatic trips

COR0020802001080C Given a specific Diesel Generator malfunction, determine the effect on any of the following: AC Electrical Distribution

Skills

262001.A2.10 Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences...: (CFR: 41.5 / 45.6) Exceeding current limitations (2.9/3.4)

Question 82   5459   (1 point(s))

Given the following:

- Reactor power is 84%
- ARI & ATWS RPT LOGIC POWER FAILURE alarm has occurred
- It has been determined that 125 VDC panel BB3 has been lost

What Alternate Rod Insertion (ARI) **SYSTEM FUNCTIONS** is/are available (if any)?

a. **NO** ARI functions are available.

b. **ALL** ARI functions are available.

c. **ONLY** control rod insertion via ARI manual initiation pushbuttons.

d. **ONLY** automatic initiation of the recirculation pump field breaker trips.

Answer 82

b. **ALL** ARI functions are available.

Foils:

a. All functions are available.
c. Automatic control rod and RPT is available.
d. Manual and automatic control rod insertion is available.

REFERENCES: COR0023302, page 18 and 21

New
Difficulty 3
Cognitive level 1

Enabling Objectives

COR0023302001060A   Predict the consequences a malfunction of the following would have on the ARI system: 125 VDC System

COR0023302001080A Briefly describe the interrelationship between ARI and the following: 125 VDC System

Skills

263000.K3.03 Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: (CFR: 41.7 / 45.4) Systems with D.C. components (i.e. valves, motors, solenoids, etc.) (3.4/3.8)

Question 83   3287   (1 point(s))

The plant is operating at 75% power when the following indications are received:

- 9-4-1/C-5, OFFGAS HIGH RAD alarm
- 9-4-1/C-4, OFFGAS TIMER INITIATED alarm
- K-1/A-4, OFFGAS FILTER HIGH D/P alarm
- OFF-gas flow indicates 100 cfm on Recorder AR-FR-47, SJAE AIR FLOW

If the above conditions are sustained for 16 minutes, what automatic actions will occur?

a. AOG-AO-901 "AOG Supply valve" closes.

b. AOG-AO-902 "AOG Return valve" closes.

c. OG-AO-254 "Offgas System Isolation valve" opens.

d. AR-AO-12 "30 Minute Holdup Pipe Drain valve" opens.

Answer 83

b. AOG-AO-902 "AOG Return valve" closes.

REFERENCE: PR 2.3.2.24 9-4-1/C-4, PR 2.4OG

Foils:

a. AOG-AO-901 remains open.
c. OG-AO-254 closes
d. AR-AO-12 closes

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0011602001080G   Describe the Off Gas system design feature(s) and/or interlock(s) that provide for the following: Automatic system isolation

COR0011602001100B   Predict the consequences of the following items on the Off Gas system: Process Radiation Monitoring failure

Skills

272000.K4.02   Knowledge of RADIATION MONITORING System design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Automatic actions to contain the radioactive release in the event that the predetermined release rates are exceeded (3.7/4.1)

Question 84   3799   (1 point(s))

Given the following conditions:

- Pressure in the reactor building is 14.7 psia.
- Pressure in the torus is 14.1 psia.
- Pressure in the drywell is 13.5 psia.

What will be the status of the primary containment vacuum relief system for these conditions?

a. **NO** vacuum breakers are open.

b. **ALL** vacuum breakers are open.

c. **ONLY** the torus to drywell vacuum breakers are open.

d. **ONLY** the reactor building to torus vacuum breakers are open.

Answer 84

b. **ALL** vacuum breakers are open.

REFERENCES: STCOR002-03-02, page 17, section II.A.4, rev. 11.

Foils:

a, c, d All vacuum breakers would be open.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

SKL012420300A030P   Given plant conditions, predict changes in the following Containment system components/parameters: Vacuum breaker-Drywell to suppression chamber (Torus)

COR0020302001230D   Predict the consequences of a malfunction of the following on the Primary containment: Drywell vacuum relief.

SKL012420300A030O Given plant conditions, predict changes in the following Containment system components/parameters: Vacuum breaker-SRV operation

Skills

223001.K4.05 Knowledge of PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Maintains proper suppression pool to drywell differential pressure (2.9/3.1)

Question 85   5471   (1 point(s))

The following conditions exist:

- Reactor power is 67% and being raised
- Throttle pressure indicates 946 psig
- Load Limit set at 839
- Flow Limiter at 110%
- Both throttle pressure transducers have failed causing the Governor and bypass valves to transfer to Manual
- THROT PRESS MONITOR light has illuminated

If a turbine trip occurs, the turbine bypass valves will . . .

a. remain closed and **CANNOT** be manually opened.

b. remain closed but can be manually opened by the operator.

c. automatically full open and **CANNOT** be manually closed.

d. automatically full open and must be manually closed.

Answer 85

d. automatically full open and must be manually closed.

At greater than 25% power with bypass valves in MANUAL, the bypass valves fail open and must be manually closed on a turbine trip.

REFERENCES: COR0020902 Student Text, 2.4DEH Attachment 4

Foils:

a. The bypass valves fail open, not closed.
b. The bypass valves fail open, not closed.
c. The valves can be manually closed.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0020902001070E Given a specific DEH Control system malfunction, determine the effect on any of the following: Bypass valves

COR0020902001080M Predict the consequences a malfunction of the following would have on DEH Control system: Low turbine inlet pressure (loss of pressure signal)

Skills

241000.K3.25 Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor cooldown (3.3/3.3)

Question 86   2116   (1 point(s))

What indications are expected when performing a coupling check on an **UN**coupled control rod?

	Overtravel Annunciator	**4-Rod Display**	**Rod Drift Red Light**
a.	ON	48	ON
b.	OFF	48	OFF
c.	ON	blank	ON
d.	OFF	blank	OFF

Answer 86

c.	ON	blank	ON

REFERENCE: Reactor Manual Control System Text, Proc. 4.3 Section 9 & 10

Foils:

a. 48 would not be displayed.
b. Would have overtravel annunciator, 48 would not be displayed and drift light would be on.
d. Would have overtravel annunciator and drift light would be on.

Bank
Difficulty 2
Cognitve Level 1

Enabling Objectives

COR0022002001090F Describe the interrelationships between RMCS and/or RPIS, and the following:
Full Core Display

Skills

214000.K1.05 Knowledge of the physical connections and/or cause- effect relationships between ROD POSITION INFORMATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Full core display: Plant-Specific (3.3/3.3)

Question 87   673   (1 point(s))

How does a loss of 125 VDC A & B affect RPS?

a. RPS logic strings A1, A2, B1 and B2 de-energized due to the TSVs closing.

b. The backup scram valves would not be available. Automatic scrams are still available.

c. RPS logic strings A1, A2, B1 and B2 de-energized due to the SDIV level transmitters losing of power.

d. The SDV drain valves close, isolating the scram discharge volume. Automatic scrams are still available.

Answer 87

b. The backup scram valves would not be available. Automatic scrams are still available.

Reference: 4.5, COR0022102

Foils:

a. Logic strings are supplied by RPS 120 VAC, TSVs would not be affected directly by the loss of 125vdc.
c. Logic strings are supplied by RPS 120 VAC, SDIV instruments would not be affected directly by the loss of 125vdc
d. SDV level instruments are supplied by RPS and would not close.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0022102001090D   Predict the consequences a malfunction of the following would have on the RPS system: D.C. electrical distribution

Skills

212000.K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: (CFR: 41.7 / 45.7) D.C. electrical distribution (2.8/3.1)

Question 88   2288   (1 point(s))

Who's approval is required for an Emergency Plan Temporary Configuration Change in accordance with procedure 3.4 (Configuration Change Control)?

a. TSC Director

b. Shift Supervisor

c. OPS/EOP Advisor

d. Engineering Team Leader

Answer 88

b. Shift Supervisor

REFERENCES: PR 3.4

Foils:

a. Must be SS or Emergency Director.
c. Must be SS or Emergency Director.
d. Must be SS or Emergency Director.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT0320109A0A010B   Discuss the following as described in Engineering Procedure 3.4, Configuration Change Control: Emergency Plan TCCs

Skills

2.2.11   Knowledge of the process for controlling temporary changes.   (CFR: 41.10 / 43.3 / 45.13)

Question 89   16474   (1 point(s))

With the plant at 100% power, HPCI inadvertently initiates. What is the **MINIMUM** required of the BOP to confirm the condition, and once confirmed, is CRS concurrence required before stopping and preventing HPCI injection?
(NOTE: The choices are listed from MINIMUM to MAXIMUM order.)

After confirming that the initiation is inadvertent on at least . . .

b. one (1) indication, obtain CRS concurrence, then stop and prevent injection.

b. one (1) indication, stop and prevent injection. CRS concurrence is **NOT** required.

c. two (2) independent indications, obtain CRS concurrence, then stop and prevent injection.

d. two (2) independent indications, stop and prevent injection. CRS concurrence is **NOT** required.

Answer 89

d. two (2) independent indications, stop and prevent injection. CRS concurrence is **NOT** required.

Two independent indications of a parameter are to be checked to validate an initiation signal. CRS permission must be obtained PRIOR to defeating an automatic initiation of an ECCS component UNLESS it is an immediate operator action.

REFERENCE: 2.0.3; 8.2.1.2, 8.2.3.1, 2.0.1.2; 7.7.1, 7.7.2, 2.4CSCS; 3.1

Foils:

a. Two independent indications of a parameter are to be checked to invalidate an initiation signal. CRS permission is NOT required.
b. Two independent indications of a parameter are to be checked to invalidate an initiation signal.
c. CRS permission is NOT required.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

OTH0151003001030C Given a video tape or during observation of crew performance determine deviation from correct behaviors, as addressed in procedure 2.0.3, Conduct of Operations, and Operations Instruction #7 for the following areas: Critical parameter control

INT032010300A010A Discuss the following as described in Conduct of Operations Procedure 2.0.1, Plant Operations Policy: Precautions and Limitations

INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

Skills

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)

Question 90   16403   (1 point(s))

During an ATWS, the RO is directed to perform alternate control rod insertion. The RO will be performing the actions to insert control rods by resetting RPS and inserting a manual reactor scram. The CRS has mandated that all peer check requirements be complied with. Another licensed operator is available to assist as necessary.

What are the peer checking requirements to perform this task?

a. All steps of the task require peer check.

b. Peer checking is waived for Panel 9-5 actions but all other steps require peer check.

c. Peer checking is waived for jumper installation but all other steps require peer check.

d. All steps of the task are waived from the peer check.

Answer 90

c. Peer checking is waived for jumper installation but all other steps require peer check.

Peer check will be performed for front panel manipulations prior to manipulating controls. This verification will be performed during steady state manipulations and whenever possible during abnormal and transient conditions. Immediate operator actions shall not be delayed to wait for peer check. Peer check can be suspended for specific tasks during transients by the CRS.

REFERENCE: OI-7, 2.0.3

Foils:

a. Back panel actions do not require peer check (jumpers).
b. Panel 9-5 actions require peer check, back panel does not.
d. Panel 9-5 actions require peer check.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT032014000A010A Discuss the following concepts as they are described in Operations Instruction #7, "Operations Expectations:" Peer check

Skills

2.1.20 Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Question 91 16633 (1 point(s))

The plant is shutdown for a refueling outage. How many ROs are required to be in the control room and what is their responsibility during the core off load?

(Assume the allowance for higher grade Licensed Operators to take the place of lower grade Licensed Operators will NOT be exercised).

a. One. Must be at the controls and also monitoring fuel movements.

b. Two. One at the controls and the other is monitoring fuel movements.

c. Two. One at the controls is monitoring fuel movements and the other for security.

d. Three. One at the controls, another is monitoring fuel movements, and a third for security.

Answer 91

b. Two. One at the controls and the other is monitoring fuel movements.

REFERENCE:

10.21; 9.2.2.4
Control Room activities affecting the transfer of fuel shall be monitored by a Licensed Operator with no other concurrent duties, who will act as the Control Room Monitor.

2.0.3; section 9, NOTE 2
Higher grade qualified Licensed Operators may take the place of lower grade Licensed or Non-licensed Operators.

2.0.3; 9.2.1
The minimum shift staffing requirements for MODE 4 or 5 shall include two active Licensed Operators (one of which must be an SRO).

2.0.3; 9.2.4.1

For this shift complement, an active Licensed SRO or RO shall be at the controls in the Control Room.

Foils:

a. Must be two. One ATC and the other dedicated to fuel movement.
c. Second RO is dedicated to fuel movements.
d. Only two are required.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT032010300C010H Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Control Room and Station Shift Staffing Requirements

Skills

2.2.30 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area / communication with fuel storage facility / systems operated from the control room in support of fueling operations / and supporting instrumentation. (CFR: 43.6)

Question 92   8875   (1 point(s))

Firefighting activities are in progress in the Reactor Building due to an ***ACTIVE*** fire. An automatic reactor scram has just occurred due to damage caused by the fire. High-high sump level conditions exist in Reactor Building Sumps B and D and the sump levels continue to rise.

What actions are required?

Start all available Reactor Building sump pumps and ...

a. cooldown the reactor at < 100 °F/hour **only**.

b. rapidly depressurize the RPV with the Main Turbine Bypass valves **only**.

c. isolate Torus Area drains **AND** rapidly depressurize the RPV with the Main Turbine Bypass valves.

d. isolate **ALL** systems that are discharging water into the affected areas **AND** cooldown the reactor at < 100 °F/hour.

Answer 92

a. cooldown the reactor at < 100 °F/hour **only**.

REFERENCE: EOP-5A.

K/A 295036 Secondary Containment Sump High Water Level

Foils:

b. No ED condition is being approached.
c. No requirement to isolate Torus area drains and NO ED condition is being approached.
d. Fire water should NOT be isolated

Bank
Difficulty 4
Cognitive Level 3

Provide to Candidate: EOP 1A and 5A with entry conditions removed

Enabling Objectives

INT00806170010600 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Skills

2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Question 93   17857   (1 point(s))

The plant is operating at 100% power when a steam line drain line breaks in the HPCI Room. This raises the temperatures in the area and a high temperature alarm is received for the SW Quad. HPCI then automatically isolates due to high temperature.

What actions are ***required*** regarding Reactor Building cooling?

a. Start **ALL** Room FCUs. Operate them until EOP-5A has been exited.

b. Start **only** the HPCI Room FCU. Operate it until EOP-5A has been exited.

c. Start **ALL** Room FCUs. Operate them until all area temperatures are below Maximum Safe Operating Value.

d. Start **only** the HPCI Room FCU. Operate it until the area temperatures is below Maximum Normal Operating Value.

Answer 93

a. Start **ALL** Room FCUs. Operate them until EOP-5A has been exited.

REFERENCE: EOP-5A

Foils:

b. All FCUs should be run.
c. The FCUs should be run until EOP-5A is exited.
d. All FCUs should be run. The RB HVAC system should not be isolated.

New
Difficulty 3
Cognitive Level 3

Provide to Candidate: EOP 5A with entry conditions removed

Enabling Objectives

INT00806170010600 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Skills

295032.EK2.02 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: (CFR: 41.7 / 45.8) Secondary containment ventilation. (3.6/3.7)

Question 94   17897   (1 point(s))

For which of the conditions below would it be appropriate to "Anticipate Emergency Depressurization" per EOP-1A, RPV CONTROL?

a. -130" (WR) and lowering RPV water level, **NO** injection available.

b. RWCU room temperature 205 °F and rising, Reactor building "A" sump 6 feet and rising.

c. 1000 psig and rising RPV pressure, 125 °F torus temperature and rising, no SRVs operable.

d. Primary Containment water level 15.5 feet and rising, injection from sources external to the primary containment required for adequate core cooling.

Answer 94

d. Primary Containment water level 15.5 feet, injection from sources external to the primary containment required for adequate core cooling.

REFERENCE: INT0080605

FOILS:

a. Step RC/L-7 does not immediately precede an ED step.
b. does not immediately precede an ED step
c. Step RC/P-5 or RC/P-6 will not require ED. Step SP/T-5, HCTL is OK.

K/A 295029 High Suppression Pool Water
New
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT00806050010500   State the criteria used to determine that "Emergency Depressurization is Anticipated".

INT00806050011000 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Skills

2.4.6 Knowledge symptom based EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Question 95   10688   (1 point(s))

- The Emergency Director declared a Site Area Emergency five (5) minutes ago.
- The TSC is **NOT** operational.
- EOP actions outside the control room are necessary to perform manual draining of the SDV.
- **ALL** Area Radiation Monitors (ARMs) on the Reactor Building 903' elevation are alarming and indicate off-scale high.
- Drywell radiation monitor RMA-RM-40A and RMA-RM-40B indicate 2 x 10^{3} rem/hour.

In addition to a TLD and PD-1, "Digital Alarming Dosimeter," which one of the following describes the **MINIMUM** additional requirements (if any) necessary to perform the directed actions per 5.8.3, "Alternate Rod Insertion Methods?" (NOTE: The choices are listed in MINIMUM to MAXIMUM order.)

a. The operator may perform the actions independently with **NO** additional radiological protection.

b. The operator must be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician **OR** the operator shall carry a survey instrument capable of monitoring radiation dose rates.

c. The operator must be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician. (An operator carrying a survey instrument capable of monitoring radiation dose rates does **NOT** satisfy the requirements.)

d. The operator may **NOT** enter Secondary Containment until the TSC is operational.

Answer 95

c. The operator must be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician. (An operator carrying a survey instrument capable of monitoring radiation dose rates does **NOT** satisfy the requirements.)

REFERENCE: ESP 5.8.3

Foils:

a. This is true if NO ARMs are in alarm.

b. This is not an option when the ARM is off-scale high.

d. This is true if the Drywell radiation monitor RMA-RM-40A or RMA-RM-40B indicate >10^4 rem/hour.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00806060010900 Identify any EOP support procedures addressed in Flowchart 6A and apply any associated special operating instructions or cautions.

Skills

295033.EK1.02 Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: (CFR: 41.8 to 41.10) Personnel protection (3.9/4.2*)

Question 96 6062 (1 point(s))

Given the following conditions:

- The plant is at 100% power.
- A Reactor shutdown is commenced at 2200 on 1/2.
- The following is the projected power reduction schedule:
  100% at 2200 on 1/2 (power reduction started)
  75% at 0800 on 1/3
  50% at 2000 on 1/3
  25% at 2200 on 1/3
  15% at 0200 on 1/4
  5 % at 0300 on 1/4

Which choice below is the **EARLIEST** time where the Primary containment oxygen concentration can exceed 4%?
(Note: Choices are listed from EARLIEST to LATEST.)

a. 2300 on 1/1/00

b. 0300 on 1/2/00

c. 2300 on 1/2/00

d. 0300 on 1/3/00

Answer 96

d. 0300 on 1/3/00

REFERENCES: COR002-03-02, page 2, rev. 11. T.S. 3.6.3.1

Foils:

a. This is the time shutdown is commenced and is > 24 hrs. prior to reducing power to < 15%.
b. This is 47 hrs. prior to reducing power to < 15%.
c. Time is 28 hrs. before and O2 must be < 4%.

K/A 223001 Primary CTMT and Auxiliaries

Bank

Difficulty 3
Cognitive Level 2

Provide to Candidate: T.S. 3.6.3.1 and bases

Enabling Objectives

INT00705070010300 Given a set of plant conditions that constitutes non-compliance with a Chapter 3.6 LCO, determine the ACTIONS that are required.

INT00705070010100 Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Skills

2.1.12 Ability to apply technical specifications for a system. (CFR: 43.2 / 43.5 / 45.3)

Question 97 3339 (1 point(s))

Given the following plant conditions:

- "A" Recirc MG set speed is lowering
- Attempts to manually control the "A" Recirc MG set have been unsuccessful
- Core Flow is 37Mlbm/hr and slowly lowering
- Core thermal power is 70% and slowly lowering

What **immediate operator action** must be performed?

a. Trip the "A" Recirculation pump.

b. Press the "A" SCOOPTUBE LOCKOUT button.

c. Scram and concurrently enter Procedure 2.1.5 "Reactor Scram."

d. Reduce the speed of "B" Recirculation pump to match the "A" Recirculation pump.

Answer 97

b. Press the "A" SCOOPTUBE LOCKOUT button.

REFERENCES: PR 2.4RR

foils:

a. Tripping "A" Recirc pump is not required under these conditions.
c. A scram is not required under these conditions.
d. Matching recirc speeds is not an immediate action and would place the plant in the instability region.

Modified
Difficulty 2
Cognitive Level 2

Provide to Candidate: ***Power to Flow Map (2.1.10)***

Enabling Objectives

INT032012400H0H00 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

INT032012400G0G00 Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Skills

295001.AA1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.7 / 45.6) Recirculation system (3.5/3.6)

Question 98   5421   (1 point(s))

Given the following conditions:

- The plant is operating at 76% power
- The A Recirc Pump trips
- Power lowers to 50% **AND** Core flow lowers to 27 x 10^6 lbm/hr
- APRM indications are oscillating 12% peak-to-peak

What action is ***required*** and what is the reason for that action?

a. Initiate actions to prevent Cold Water Stratification.

b. Initiate a manual Scram to terminate power oscillations.

c. Raise Recirc flow to exit the Stability Exclusion Region.

d. Reduce B Recirc Pump speed to prevent Jet Pump Vibrations.

Answer 98

c. Raise Recirc flow to exit the Stability Exclusion Region.

REFERENCE: PR 2.4.RR; PR 2.1.10

Foils:

a. Stratification is not a concern if Core flow >20%.
b. Scram is not directed by procedure.
d. Jet Pump vibrations are not a concern here. Lowering flow will aggravate Power to Flow situation.

K/A 202002 Recirculation Flow Control

Modified
Difficulty 3
Cognitive Level 3

Provide to Candidate: **Power to Flow Map (2.1.10)**

Enabling Objectives

INT032012400G0G00 Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

INT032012400I0I00 Given plant condition(s) and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

INT032012400H0H00 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

2.1.25 Ability to obtain and interpret station reference materials such as graphs / monographs / and tables which contain performance data. (CFR: 41.10 / 43.5 / 45.12)

Question 99   17864   (1 point(s))

While using the 'B" loop of RHR to cool the suppression pool, RHR-MO-66B, B HX BYPASS VLV must be fully CLOSED.

Which one of the following precautions apply and what is the bases for this precaution?

a. Do **NOT** hold the control switch in close after the green indicating light turns on to prevent hammering the valve.

b. Do **NOT** hold the control switch in close longer than 1 second after the green indicating light turns on to prevent tripping the valve breaker.

c. Hold the control switch in close for 5 seconds after the red indicating light turns off to ensure the valve closure is terminated by torque switch.

d. Hold the control switch in close for 3 seconds after the red indicating light turns off to ensure the valve closure is terminated by the closed limit switch.

Answer 99

c. Hold the control switch in close for 5 seconds after the green indicating light turns on to ensure the valve closure is terminated by torque switch.

RHR-MO-66B, B HX BYPASS VLV is a throttle valve and throttle valves are held in closed an additional 5 seconds to ensure they are closed by their torque switch.

Reference: 2.2.69.3, Section 2.6

Foils:

a. This is based on the precaution from the procedure on seal-in limitorque operators.
b. This is based on the precaution from the procedure on seal-in limitorque operators.
d. The valve is held for 5 seconds (3 seconds is time delay for reversing directions) to prevent terminating the closure with the close limit switch.

K/A 219000 RHR/LPCI: Torus/Pool Cooling Mode

New
Difficulty 2
Cognitive Level 1

Enabling Objectives

SKL012422300A0200 Explain the Residual Heat Removal system limitations and precautions as stated in the SOP 2.2.69, SOP 2.2.69.1, SOP 2.2.69.2 and SOP 2.2.69.3.

COR0022302001050B Briefly describe the following concepts as they apply to the RHR system: Valve operation

Skills

2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

Question 100 5226 (1 point(s))

Given the following conditions:

- The plant is operating at 100% power with a Nitrogen Makeup to the Primary Containment in progress
- A loss of the "B" Reactor Protection System (RPS) Motor Generator Set occurs
- Torus N2 Supply Isolation Valve MO-1301 **AND** Drywell N2 Supply Isolation Valve MO-1312 closed
- RPS power has been restored

Which of the following actions are **required** to restore the Nitrogen makeup flowpath for these conditions?

MO-1301 **AND** MO-1312 switches ___________ when the group isolation signal is reset.

After the isolation has been reset, MO-1301 **AND** MO-1312 switches must be positioned to ______________ to re-open the valves.

a. **must** be in OVRD
OPEN

b. **must** be in CLOSE
OPEN

c. **can** be in any position
CLOSE **AND THEN** to OPEN

d. **can** be in any position
OVRD **AND THEN** to OPEN

Answer 100

c. **can** be in any position
CLOSE **AND THEN** to OPEN

EXPLANATION OF ANSWER:
After the isolation is reset, the control switches are positioned to close to reset the valve circuit and then to open to reposition the valve.

REFERENCE: STCOR0020302 Containment Rev 11; PR 2.1.22 Recovering From a Group Isolation Page 14 Section 8

Foils:

a. The isolation can be reset with the switches in any position. The switch must be left in OVRD to keep the valve open for this choice.
b. The isolation can be reset with the switches in open.
d. The valves will not open unless the switch is first positioned to close and then to open.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0020302001130E Describe the PCIS design features and/or interlocks that provide for the following: Operator action to defeat/reset isolations

Skills

223002.A4.03 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Reset system isolations (3.6/3.5)

Examination Cover Sheet

Exam ID: 2002 SRONR **Total Points:** 100

Exam Date: 08/05/2002

Description: August 2002 SRO NRC Written Exam

Student Name:
__________________________________ **Date**: ____________ **Grade:** ____________

Graded By:
__________________________________ **Date:** ____________

Approved By:
__________________________________ **Date:** ____________

Reviewed By Examinee:
____

Question 1 17898 (1 point(s))

The plant is operating at 100% power with a normal electric plant lineup when an oil leak occurs on the Normal Tranformer. The oil catches on fire. The fire brigide responds and attempt to extinguish the blaze. The oil and the fire is contained to the Normal Transformer area.

What is the potential effect of this fire?

a. The loss of one (1) Recirculation pump **only**.

b. A main generator trip AND Reactor scram **only**.

c. The loss of **BOTH** Recirculation pumps AND main generator trip AND Reactor scram.

d. The loss of **only** one (1) Recirculation pump AND main generator trip AND Reactor scram.

Answer 1

d. The loss of **only** one (1) Recirculation pump AND main generator trip AND Reactor scram.

The NORMAL transformer will trip eventually due to the fire.

REFERENCE: **C-2/A-8,** NORMAL TRANSFORMER LOCKOUT, COR0010102

Foils:

a. The main generator will trip as well.
b. A recirc pump will also trip.
c. Only one recirc pump will trip.

New
Difficulty 3
Cognitive Level 3

Enabling Objectives

INT0320134E0E0100 Given plant condition(s), determine from memory any automatic actions listed in the applicable Abnormal/Emergency Procedure(s) which will occur due to the event(s).

COR0010102001080A   Predict the consequences of the following on plant operation: Loss of Normal and Startup transformers

Skills

600000.AA2.04 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: The fire's extent of potential operational damage to plant equipment (2.8/3.1)

Question 2   17868   (1 point(s))

What is the effect of losing 125 VDC to the turbine generator?

a. Loss of DEH pressure.

b. Loss of main turbine remote trip capability.

c. DEH shifts to the standby pressure regulator.

d. The generator field breaker automatically opens.

Answer 2

b. Loss of main turbine remote trip capability.

Reference: 5.3DC125

Foils:

a. DEH pressure is not affected. The pumps are AC powered.
c. They are not affected by the power loss.
d. The generator field breaker must be manually opened.

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

COR0020702001080C   Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Systems with DC components (i.e., valves, motors, solenoids, etc.)

COR0011302001030E   Describe the interrelationships between Main Turbine Generator and Auxiliaries and the following: DC electrical distribution

Skills

245000.K1.09 Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) D. C . electrical distribution (2.7/2.7)

Question 3   17847   (1 point(s))

Following a a reactor scram from 100% power the main turbine generator has automatically tripped.

Which one of the following groups of breakers will the generator trip automatically open?

The main generator output breakers, the generator field breaker and the feeder breakers from . . .

a. the auto transformer.

b. the startup transformer.

c. the normal transformer.

d. busses 1A & 1B to busses 1F & 1G.

Answer 3

c. the normal transformer.

Reference: COR001-13-01, Section III Instrumentation and Controls.

Foils:

a. The generator field breaker is tripped and the startup breakers close vice open.
b. The generator field breaker is tripped.
c. The feeder breakers from the startup transformer will close vice open.

New
Difficulty: 2
Cognitive level:1

Enabling Objectives

COR0010102001130A   Predict the consequences of the following events on the AC Electrical Distribution System: Turbine/generator trip

Skills

295005.AA1.07 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.7 / 45.6) A.C. electrical distribution (3.3/3.3)

Question 4 17889 (1 point(s))

Why is Recirculation pump speed limited following a reactor scram?

a. To protect the Reactor Recirculation pump from damage due to lack of NPSH.

b. To prevent damage to the Recirculation pump thrust bearing due to inadequate film layer.

c. To protect the Reactor Recirculation pump motor from overload due to fluid density increase.

d. To prevent damage to the Reactor Recirculation pump seals caused by excessive differential pressure.

Answer 4

a. To protect the Reactor Recirculation pump from damage due to lack of NPSH.

REFERENCE: COR0022202

Foils:

b. Thrust bearing damage is not a reason for the #1 limiter.
c. Motor overload is not a reason for the #1 limiter.
d. Seal damage is a concern for the #1 limiter only if the Discharge valve is closed.

New
Difficulty: 2
Cognitive level:1

Enabling Objectives

COR0022202001100A Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following: Adequate Recirculation Pump NPSH

Skills

295006.AK3.06 Knowledge of the reasons for the following responses as they apply to SCRAM: (CFR: 41.5 / 45.6) Recirculation pump speed reduction: Plant-Specific (3.2/3.3)

Question 5   17859   (1 point(s))

The plant is operating at 30% when the reactor operator is directed to raise power using Recirculation flow. As the controller output is raised, a momentary (1.5 seconds) loss of signal occurs from the "A" Recirculation Flow Controller. The operator continues to raise the controller output for several more seconds.

How will the "A" Recirculation MG Set be affected by this momentary loss and operator action?

a.	Automatically run back to ~ 22% speed

b.	A scoop tube lockup will prevent any further speed change.

c.	After a 1.5 second pause recirculation pump speed will rise as directed.

d.	Speed will initially rise, then lower rapidly for 1.5 seconds then rise again for several seconds.

Answer 5

b.	A scoop tube lockup will prevent any further speed change.

Reference: 2.2.68, Attachment 1 Section 2.1.6

Foils:
a.	There is no runback.
b.	Scoop tube lockup prevents any speed changes.
d.	Scoop tube lockup prevents any speed changes.

New
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0022202001130F   Given plant conditions, determine if any of the following should occur: Recirculation MG set scoop tube lock.

COR0022201001060A   Given plant and/or reactor recirculation system conditions, apply the design features and/or interlocks that provide for the following: MG Set Scoop Tube Lockout

SKL012422200A030I Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: RR pump speed

Skills

202002.K1.05 Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
Recirculation MG set: Plant-Specific (3.5/3.5)

Question 6   16478   (1 point(s))

The plant is at 20% power when the following annunciator alarms:

- 9-4-3 / A-3, RECIRC PUMP A SEAL TROUBLE
- RONAN (1820): RECIRC PUMP A SEAL STAGING FLOW HIGH

The following indications are observed:

- Seal cavity #1 (inner) pressure 950 psig
- Seal cavity #2 (outer) pressure 950 psig

What is the state of the A Recirculation Pump seals?

a. **Only** seal #1 is failed.

b. **Only** seal #2 is failed.

c. **Both** #1 and #2 seals are failed.

d. The seal staging flow orifice is clogged.

Answer 6

a. Only seal #1 is failed.

Equal pressure between RR-PI-32A (RR-PI-32B)SEAL 2 BRG CAV and RR-PI-33A (RR-PI-33B) SEAL 1 BRG CAV indicates a #1 seal failure.
Normally seal cavity 2 pressure should be about 1/2 of cavity 1 pressure.

REEFERENCE: 2.3_9.4.3
2.4RR; Attachment 3 - 1.3

Foils:

b. Very low pressure in seal cavity #2, indicated on RR-PI-32A (RR-PI-32B) SEAL 2 BRG CAV, indicates #2 seal failure. Upper cavity pressure will approach lower cavity pressure.
c. The pressure relationship between the seals would remain (1/2 for pressure). Would also receive (1819) Recirc Pump A Outer Seal Leakage Flow High Alarm.
d. Will not receive seal staging flow high alarm for this condition.

Bank
Difficulty 2
Cognitve Level 2

Enabling Objectives

COR0022202001140A Given plant conditions determine if the following has occurred: #1 seal failure only.

Skills

202001.A4.11 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Seal pressures: Plant-Specific (3.2/3.3)

Question 7   5497   (1 point(s))

Given the following conditions:

- The generator has been synchronized to the grid
- Bypass Valves are closed
- The load demand remains at 200
- Reactor power is raised using reactor recirculation flow

How will the bypass valves and DEH respond?

As reactor power is raised, turbine bypass valves . . .

a. remain closed **and** DEH remains in MODE 3.

b. remain closed **and** DEH remains in MODE 4.

c. open when load exceeds 200 MW **and** DEH transfers to MODE 3.

d. open when load exceeds 200 MW **but** DEH remains in MODE 4.

Answer 7

c. open when load exceeds 200 MW **and** DEH transfers to MODE 3.

EXPLANATION OF ANSWER: The load control signal enters a low value gate with pressure control signal. Whichever is lower will be passed through. Until megawatts exceeds 200 the load signal will be lower than the pressure control signal. At this point the opening of the control valves will be limited. This will cause the bypass valves to open. When the bypass valves open DEH will transfer to MODE 3.

REFERENCES: COR002-09-02, DEH. PR 2.2.77.1, DEH Control System

Foils:
a. Bypass valves will open and DEH was in Mode 4.
b. Bypass valves will open and DEH transfers to Mode 3.
d. DEH will transfer to Mode 3.

Bank
Difficulty: 2

Cognitive Level: 2

Enabling Objectives

COR0020902001040L Describe how the DEH control system operates to control the following: Bypass valve position

COR00209020011100 Define the four DEH Control system operating Modes used at CNS.

COR00209020011200 Given a specific DEH Control system operating Mode, identify the parameter controlled and components positioned for controlling it.

Skills

295007.AK2.01 Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: (CFR: 41.7 / 45.8) Reactor/turbine pressure regulating system (3.5/3.7)

Question 8   17890   (1 point(s))

During a plant transient, SRV "D" lifts in response to high pressure. When a tailpipe vacuum breaker operates to equalize pressure, it sticks in the **OPEN** position.

What are the possible consequences of this failure of the SRV tailpipe vacuum breaker?

If the SRV lifts again, . . .

a. the pressure trapped in the discharge pipe would raise the operating setpoint of the SRV, causing possible damage to the Reactor Vessel and piping.

b. the water that was drawn up into the discharge pipe would cause a water hammer to occur, causing possible damage to the SRV, piping, and torus.

c. the steam passing through the SRV will be released directly into the torus airspace, bypassing the pressure suppression function of the primary containment.

d. the steam passing through the SRV will be released directly to the drywell atmosphere, producing primary containment conditions similar to a small break LOCA.

Answer 8

d. the steam passing through the SRV will be released directly to the drywell atmosphere, producing primary containment conditions similar to a small break LOCA.

REFERENCE: P&ID 2028

Foils:

a. This will not occur if a vacuum breaker sticks open.
b. This is the effect if the vacuum breakers stick closed.
c. The SRV tailpipe vacuum breakers relieve to the drywell, not torus airspace.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0021602001080H   Predict the consequences a malfunction of the following would have on the NPR system: Discharge line vacuum breaker

Skills

239002.A2.01   Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...:   (CFR: 41.5 / 45.6) Stuck open vacuum breakers (3.0/3.3)

Question 9   1499   (1 point(s))

An SRV is opened during normal power operations.

When read on MS-TR-166, SAFETY AND RELIEF VALVE LEAKAGE TEMPS, tail pipe temperature will indicate . . .

a. between 250°F and 280°F.

b. between 285°F and 325°F.

c. between 385°F and 400°F.

d. between 500°F and 550°F.

Answer 9

b. between 285°F and 325°F.

REFERENCE: COR002-16-02, page 23, section IV.E.2, rev. 09, Steam Tables

Foils:

a. Saturation temp for 30 psia.
c. Numbers to round out spacing of other numbers.
d. Saturation temp of vessel.

Bank
Difficulty 2
Cognitive Level 2

Provide to Candidate: Steam Tables

Enabling Objectives

SKL012421600A030F Given plant conditions, predict changes in the following NPR system components/parameters: SRV tailpipe temperatures

COR0021602001040A   Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Tail pipe temperatures

Skills

218000.A4.06   Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) ADS valve tail pipe temperature (3.5/3.6)

Question 10   12692   (1 point(s))

What is the reason for the decrease in MCPR associated with the "Feedwater Controller Failure (Maximum Demand)" transient?

a. Fuel heating due to loss of cooling

b. Fuel heating due to increased pressure

c. Positive reactivity due to void collapse

d. Positive reactivity from colder moderator

Answer 10

d. Positive reactivity from colder moderator

Reference: INT0060119 Student Text p. 33

Foils:
a.b.c. The loss of feedwater heating causes a reduction in moderator temperature and increases core inlet subcooling.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT00601190010400   Given an Anticipated Operational Transient and a list of reasons, select the correct response why the given transient would have MCPR limitations.

INT00601140010400   Given a transient and list of reasons, choose the reason the given transient would have MCPR limitations.

Skills

295014.AK2.06 Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) Moderator temperature. (3.4/3.5)

Question 11   17862   (1 point(s))

A station startup is in progress, reactor temperature is 178°F and rising at 80°F/hr. The Reactor Water Cleanup system (RWCU) is being used to control RPV water level when a high NRHX outlet temperature isolates RWCU.

How is RPV water level affected and what is the effect on the plant startup?

RPV water level . . .

a. rises. The startup may continue.

b. lowers. The startup may continue.

c. rises. The startup must be stopped due to inability to control RPV water level.

d. lowers. The startup must be stopped due to inability to control RPV water level.

Answer 11

c. rises. The startup must be stopped due to inability to control RPV water level.

Reference: 2.4RXLVL

Foils:
a.b.d. The loss of RWCU will prevent its use for RPV water level control (draining excess water via RWCU blowdown. Per 2.4RXLVL any power change must be stopped.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

SKL012412000A030A   Given plant conditions, predict changes in the following RWCU system components/parameters: Reactor vessel water level.

COR0012001001110D   State how the following systems interrelate with the operation of the RWCU System: Main Condenser

Skills

204000.K3.02   Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor water level (3.1/3.1)

Question 12   17863   (1 point(s))

12 hours after shutdown, a loss of shutdown cooling has occurred. The crew has entered abnormal procedure 2.4SDC, "Shutdown Cooling Abnormal."

Which of the following systems can be used for alternate decay heat removal?

a. Fuel Pool Cooling

b. Reactor Water Cleanup

c. Turbine Equipment Cooling

d. Diesel Ventilation Cooling Towers

Answer 12

b. Reactor Water Cleanup

RWCU can be lined up per 2.2.66 for alternate decay heat removal.

Referernce: 2.4SDC

Foils:

a. Fuel pool cooling will not be available until the reactor head has been removed and the cavity flooded and the gates removed this will not occur within 12 hours of shutdown.
c. TEC is not an alternate decay heat removal system.
d. Diesel Ventilation Cooling Towers are not alternate decay heat removal systems.

New
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0012002001050C   Briefly describe RWCU operation under the following conditions: Loss of Shutdown Cooling

Skills

205000.K5.03 Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.5 / 45.3)
Heat removal mechanisms (2.8/3.1)

Question 13   5030   (1 point(s))

Given the following conditions:

- The Reactor Core Isolation Cooling (RCIC) System is operating in the Test Mode.
- The RCIC flow controller is in BAL.
- The RCIC TURB TEST Switch is in TURB TEST.
- The RCIC TURB TEST POWER Switch is ON.
- RCIC flow is 300 gpm and turbine speed is 2500 rpm (indicated in the control room).

The operator then jogs MO-30, RCIC Test Bypass To ECST Valve, in the CLOSED direction. MO-30 is still partially open.

What is the expected ***final*** response of RCIC turbine speed and system flow if all equipment operates as designed?

a. RCIC turbine speed rises.
   System flow rises.

b. RCIC turbine speed lowers.
   System flow lowers.

c. RCIC turbine speed is unchanged.
   System flow lowers.

d. RCIC turbine speed is unchanged.
   System flow rises.

Answer 13

c. RCIC turbine speed is unchanged.
   System flow lowers.

EXPLANATION OF ANSWER: With the Test Controls ON and no initiation signal present, the Flow Controller is isolated from the turbine governor and turbine speed is controlled with the Test Pot. Positioning MO-30 and adjusting turbine speed regulates flow. As MO-30 is jogged in the closed direction, flow will lower and turbine speed will remain constant.

REFERENCE: STCOR0021802 Section III.C
PR 2.2.67.1; Section 8.5
DWG 971E264 Sh. 5

Foils:

a. Speed remains constant, flow lowers
b. Speed remains constant.
d. Flow lowers.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0021802001100H Predict the consequences of the following on the RCIC system: System Valve closures and openings

Skills

217000.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: (CFR: 41.5 / 45.5) RCIC flow (3.7/3.7)

Question 14   17891   (1 point(s))

The plant was operating at full power with RCIC tagged out when it scrammed due to trip of both Feedwater pumps. HPCI injecting at full flow recovered RPV water level from - 50" (WR) to +35" (WR). HPCI was tripped as RPV water level reached + 35" (WR). Conditions are currently:

-	Reactor pressure	700 psig (rising at 10 psig per minute)
-	MSIVs	Open
-	Time after scram	5 minutes
-	CRD pumps	both tripped
-	Drywell pressure	0.3 psig

If <u>no additional operator actions are taken</u>, what is the expected RPV water level response over the next 10 minutes and why?

RPV water level will . . .

a. lower below the low level alarm point due to shrink.

b. lower below the low level alarm point due to steam loads reducing RPV water inventory.

c. rise above the high RPV water level trip setpoint due to swell.

d. rise above the high RPV water level trip setpoint due to Startup Valve leakage exceeding decay heat requirements.

Answer 14

c. rise above the high RPV water level trip setpoint due to swell.

The specific volume change from ECST water to saturated liquid at 700 psig results in ~ 40% increase in gallons/inches of RPV level. 85" of cold water added = ~ 119" (34" more or + 69")

REFERENCE: Plant events, Steam Tables

Foils:

a. RPV water level will rise. Level will not shrink.

b. RPV water level will rise. BPVs will be closed at this pressure and remaining steam loads will not lower level under these conditions.

d. Rx pressure is above Condensate Booster pump discharge pressure.

New
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0021102001100F Predict the consequences of the following on the HPCI system: Low reactor water level

SKL012421100A030C Given plant conditions, predict changes in the following HPCI system components/parameters: Reactor water level

Skills

295008.AA2.05 Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Swell. (2.9/3.1)

Question 15   12444   (1 point(s))

The plant was operating at power when a loss of off-site power occurred. The reactor scrammed and HPCI started on low reactor water level. Reactor water level quickly recovered and the HPCI turbine tripped on high RPV water level. The following plant conditions were present:

- Reactor water level   45" (NR) (lowering slowly)
- Reactor pressure   850 psig (rising slowly)
- Drywell pressure   2.2 psig (rising slowly)

What is/are the **MINIMUM** action(s) required to restart HPCI ***at this time***?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

b. Momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

c. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

d. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

Answer 15

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

During the transient drywell pressure has risen to greater than the initiation setpoint for HPCI. Since an automatic initiation signal is present, if the operator depresses the Reactor Hi Water Level Signal Reset pushbutton the system will reinitiate.

REFERENCE: 2.2.33

Foils:
b, c, d - only the high level trip reset need be depressed.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0021102001120A Given plant conditions, determine if the following HPCI actions should occur: System initiation

SKL012421100A030J Given plant conditions, predict changes in the following HPCI system components/parameters: Turbine speed

COR0021102001100V Predict the consequences of the following on the HPCI system: High reactor water level

Skills

206000.K4.07 Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Automatic system initiation: BWR-2,3,4 (4.3*/4.3)

Question 16   3790   (1 point(s))

Service Water is cross-connected to REC from the Control Room. SW is supplying cooling to the critical loops.

How are REC/Service Water systems affected (if at all) by a Group 6 isolation signal?

a. Service Water will be isolated.

b. Service Water **AND** REC will **NOT** be affected.

c. REC will automatically realign to the critical loops.

d. Some Service Water flow will bypass the critical loops.

Answer 16

d. Some Service Water flow will bypass the critical loops.

REFERENCES: SOP 2.2.65.1

Foils:

a. Service Water will NOT isolate.
b. Both Service Water and REC are affected.
c. REC will not automatically realign.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

SKL012421900A030L   Given plant conditions, predict changes in the following Reactor Equipment Cooling System components/parameters: SW/REC Crosstie

COR0021902001040B   Describe the REC design features and/or interlocks that provide for the following: Service Water Crosstie to REC

COR0021902001040D   Describe the REC design features and/or interlocks that provide for the following: Isolation of Non-Critical Cooling loops

COR0021902001110A   Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation

Skills

295018.AA1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.7 / 45.6) Backup systems (3.3/3.4)

Question 17   16475   (1 point(s))

With the plant at 65% power, an attempt is made to single **notch** withdraw a control rod from notch 24 to notch 26. While the control rod is being withdrawn, a malfunction of the RMCS timer causes a constant withdrawal signal to be sent to the selected control rod.

Assume NO additional operator actions and that all control rods move at nominal control rod speed.

What is the FINAL POSITION of the control rod?

a. Notch 26

b. Notch 28

c. Notch 30

d. Notch 48

Answer 17

b. Notch 28

Timer malfunction deselects the rod after 2 seconds (which is ½ second longer than the normal timer), causing the control rod to be de-selected

REFERENCE: IOP 4.3; Attachment 3, 1.3.2

Foils:
a. From the given power level an Rx scram will not occur.
c. An extra ½ second will not result in a two (2) notch change.
d. Rod will be de-selected after 2 seconds of motion.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0022002001040A   Describe the RMCS design features and/or interlocks that provide for the following: Detection of sequence timer malfunction

COR0022002001060A Given a RMCS control manipulation, predict and explain the response of the following: Control rod position

COR0022002001080A Given a specific RMCS and/or RPIS malfunction, determine the effect on any of the following: Ability to move control rods

Skills

201002.A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Control rod position (3.4/3.3)

Question 18   5006   (1 point(s))

Given the following conditions:

- A plant startup is in progress with power at 8%
- Group 42 rods are being withdrawn
- ONLY one rod remains to be withdrawn in Group 42
- All other control rods are at their proper position
- The operator attempts to select AND withdraw a rod in Group 43 one notch

What is the response of the Group 43 control rod?

The Control rod will . . .

a. **NOT** withdraw and withdrawal blocks will be applied.

b. **NOT** select and **NO** withdrawal blocks will be applied.

c. withdraw one notch, then withdrawal blocks will be applied.

d. withdraw one notch and **NO** withdrawal blocks will be applied.

Answer 18

d. withdraw one notch and **NO** withdrawal blocks will be applied.

EXPLANATION OF ANSWER: RWM will latch to group 43 when the first rod moves off of its insert limit. This will leave the group 42 rod as an insert error. No rod blocks will result from one insert error.

REFERENCE: PR 4.2 Rod Worth Minimizer

Foils:

a. Rod will withdraw and no blocks are applied.
b. Nothing prevents selecting the rod in this condition.
c. no rod blocks will result.

Bank

Difficulty 4
Cognitive Level 2

Enabling Objectives

COR0022602001040C Describe the RWM design features and/or interlocks that provide for the following: Withdraw blocks and errors

COR0022602001050I Briefly describe the following concepts as they apply to the RWM: Withdraw block

COR0022602001050C Briefly describe the following concepts as they apply to the RWM: Latched groups

COR0022602001080B Predict the consequences of the following items on the RWM: Out of sequence rod movement

Skills

201006.K4.01 Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Insert blocks/errors: P-Spec (Not-BWR6) (3.4/3.5)

Question 19 17865 (1 point(s))

A LOCA is in progress with drywell spray in service on the "A" RHR loop. "B" RHR loop is in torus cooling with suppression pool temperature 78°F. Due to a logic failure, the RHR drywell spray valves will not automatically close.

What is a possible consequence if the Reactor Building to Torus Vacuum Breakers fail CLOSED under these conditions?

a. Implosion of the primary containment.

b. Over-pressurization of the primary containment.

c. Excessive differential pressure from the Drywell to the Torus.

d. Excessive differential pressure from the Torus to the Drywell.

Answer 19

a. Implosion of the primary containment.

The Reactor Building to Torus Vacuum Breakers open on a negative pressure in the torus to protect against external over-pressurization which is caused by RHR condensing the steam in the torus free air space.

Reference: Tech. Spec Bases section B.3.6.1.7

Foils:

b. This failure would not result in containment overprssure.
c. The downcomers prevent this pressure exceeding ~ 2 psid.
d. Torus to drywell vacuum breakers would open on a high internal pressure.

New
Difficulty 2
Cognitive Level 3

Enabling Objectives

COR0022302001080L Predict the consequences a malfunction of the following will have on the RHR system: Reactor building to Suppression chamber vacuum breakers

COR0020302001230D Predict the consequences of a malfunction of the following on the Primary containment: Drywell vacuum relief.

COR0022302001040H Describe the interrelationship between the RHR system and the following: Primary Containment

Skills

230000.K6.09 Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE: (CFR: 41.7 / 45.7) Reactor building to suppression pool vacuum breakers (3.5/3.8)

Question 20   9228   (1 point(s))

The plant was at 100% power when a large break LOCA occurred. Reactor water level initially lowered to +5" (corrected Fuel Zone), then was restored using RHR pumps which automatically started. To control the level rise, the BOP stopped RHR pumps A and D by momentarily placing the control switches to STOP.

Before any additional actions were taken, a loss of off-site power (LOOP) occurred.

Which RHR pumps will be operating ten (10) seconds after the DGs energize the emergency buses, if any?

a. All RHR pumps remain OFF.

b. All RHR pumps are running.

c. RHR pumps B and C are running. Other pumps remain OFF.

d. RHR pumps A and D are running. Other pumps remain OFF.

Answer 20

c. RHR pumps B and C are running. Other pumps remain OFF.

A and D RHR pumps are OFF with a STOP signal sealed in. Even if power is lost and restored the STOP signal for A and D RHR pumps remains sealed in. B and C RHR pumps will start 5 seconds after power is restored to their respective bus.

REFERENCE: 2.2.69; Attachment 1 - 2.1.1.2.a, 2.1.1.2.b, 2.1.2.2

Foils:

a. B and C pumps start after 5 seconds.
b. A and D RHR pumps remain OFF.
d. A and D pumps remain OFF. B and C RHR pumps will be running.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0022302001130D State the purpose of the following items related to the RHR system: Pumps

COR0022302001030F Describe RHR System design feature(s) and/or interlocks which provide for the following: Emergency Diesel Generator load sequencing

COR0022302001060I Given an RHR control manipulation, predict and explain changes in the following: Emergency Diesel Generator loading

COR0022302001150B Given plant conditions, determine if the following should occur: RHR pump start

Skills

203000.K2.01 Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (3.5*/3.5*)

Question 21 1729 (1 point(s))

The RPV is at 30 psig operating in Shutdown Cooling using the "B" RHR pump.

The SDC Outboard Isolation Valve, MO-17, will. . .

a. close if Reactor pressure is above 72 psig or a PCIS Group 2 isolation signal is received.

b. remain "as is" under all conditions, because it is maintained de-energized once Shutdown Cooling is established.

c. close if a PCIS Group 2 isolation signal is received, but will remain open if Reactor pressure rises above 72 psig.

d. remain "as is" under all conditions, because the Shutdown Cooling Isolation logic is disabled below 72 psig Reactor pressure.

Answer 21

a. close if Reactor pressure is above 72 psig **OR** a PCIS Group 2 isolation signal is received.

REFERENCE: RHR Text, Procedure 2.2.69, Attach. 1, step 2.2.6.3

Foils:

b. The valve is not allowed to be de-energized until Mode 4 is reached.
c. High Rx pressure will close the SDC isolation valve.
d. The isolation logic is never disabled, even in EOPs.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0022302001030B Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of piping over-pressurization

COR0022302001030J Interpret RHR System design feature(s) and/or interlocks which provide for the following: High pressure isolation

COR0022302001040L Describe the interrelationship between the RHR system and the following: PCIS

Skills

223002.K1.08 Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Shutdown cooling system/RHR (3/4/3.5)

Question 22   1744   (1 point(s))

Following a LOCA, the following conditions are present :

- Reactor pressure   700 psig
- RPV water level   - 100 in (wide range)
- Drywell pressure   11.0 psig
- Torus water temp   104 °F

What are the **MINIMUM** actions that are required in order to initiate Drywell Sprays with the "A" RHR loop? (NOTE: The choices are listed in MIMIMUM to MAXIMUM order.)

a. Place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

b. Depress Containment Spray Initiation Signal Reset pushbuttons, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

c. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

d. Place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

Answer 22

c. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

REFERENCE: RHR Text (COR0022302), Procedure 2.2.69, Attach. 1, Note page 43 & 2.2.10.2

Foils:

a. Containment Cooling Valve Control Permissive switches must be placed in MANUAL
b. There is no need to reset the containment spray initiation.
d. There is no need to override 2/3 core height.

Bank
Difficulty 2
cognitive Level 2

Enabling Objectives

COR0022302001030P Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling

COR0022302001170C Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Skills

226001.A1.05 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: (CFR: 41.5 / 45.5) System lineup (3.1/3.4)

Question 23 16499 (1 point(s))

During operation at full power annunciator 9-3-3/A-5, CORE SPRAY B BREAK DETECTION is received.

NO other annunciators alarm. A station operator sent to the d/p indicating switch reports that the d/p is +4.0 psid.

What is the significance of this alarm and d/p indication on core spray flow during a subsequent Core Spray initiation?

Core spray flow will flow …

a. into the Drywell through the broken pipe.

b. inside the core shroud and out the broken pipe.

c. into secondary containment through the broken pipe.

d. into the downcomer region of the reactor through the broken pipe.

Answer 23

d. into the downcomer region of the reactor through the broken pipe.

REFERENCE: 2.3_9-3-3, ST COR002-06-02

Justification: The alarm and d/p reading indicate the break is outside the shroud but inside the reactor.

Foils:

a. The indicated d/p would be pegged high (+1000 psig).
b. The indicated d/p would be low -3.5 psig.
c. The instrument measures d/p downstream of the check valve inside the primary containment.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0020602001090C   Predict the consequences of the following items on the Core Spray System:
Core Spray line break

Skills

209001.A4.11   Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
System flow (3.7/3.6)

Question 24 17871 (1 point(s))

The 12.5KV System has been removed from service. What Fire Protection System Fire Pumps (C, D, E) are available for use 15 minutes after the loss of 12.5KV?

a. **Only** C

b. **Only** D

c. C **and** D

d. D **and** E

Answer 24

c. C **and** D

C Fire Pump is powered from MCC-E and the Diesel pump ("D") will start from its batteries.

Reference: 2.2.90: 6.3, 6.26,6.27, and 6.28, COR0010501

Foils:
a. D will start as well
b. C will start as well
d. E pump has lost power

New
Difficulty 2
Cognitive level 1

Enabling Objectives

COR0010502001060A State the electrical power supply to the following: Diesel Fire Pump starting circuitry

COR0010502001060B State the electrical power supply to the following: Electric Fire Pumps "C" & "E"

Skills

286000.K2.02 Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (2.9/3.1)

Question 25   16470   (1 point(s))

Given the following conditions:

- The piping just downstream of the 'A' SBLC squib valve (SLC-14A) is completely obstructed.

- The control switch for SLC PUMP A on Panel 9-5 is taken to the START position, then to the STOP position when the no-flow condition is recognized.

- The control switch for SLC PUMP B on Panel 9-5 is taken to the START position. The SLC Pump B trips. Then the control switch is taken to the STOP position.

Which one of the following describes the **MINIMUM** steps necessary to inject SLC under the current conditions? (Note: The choices are listed in MINIMUM to MAXIMUM order.)

a. Return the control switch for SLC PUMP A on Panel 9-5 to the START position.

b. Place the control switches for SLC PUMP A **AND** SLC PUMP B on Panel 9-5 to START.

c. Start the B SLC Pump using the local test station keylock switch.

d. Perform alternate boron injection with any available system per EOP-5.8.8.

Answer 25

a. Return the control switch for SLC PUMP A on Panel 9-5 to the START position.

Justification: Either pump can discharge through either squib valve once both switches are placed in START. the 'B' squib valve fires when SLC PUMP B is started, when A is re-started it will discharge through the B squib valve.

REFERENCE: 2.2.74; Attachment 1.2.2, 1.3.1, 1.3.2, 1.3.3, 1.3.4

foils:

b. No need to place the SLC PUMP B control switch in start the squib valve has already been fired.
c. The breaker has tripped this would not start the pump.
d. No need to take the time for alternate injection. A SLC pump can inject.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0022902001060D Briefly describe the following concepts as they apply to SLC system: Squib valve operation

COR0022902001080D Given a SLC component manipulation, predict and explain the changes in the following: Flow indication

Skills

211000.K5.04 Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: (CFR: 41.5 / 45.3) Explosive valve operation (3.1/3.2)

Question 26   16625   (1 point(s))

The plant is at 100% power with TIP traces in progress. Only "A" TIP machine is being used at this time. Currently, TIP "A" has reached the Core Bottom Limit and is moving at slow speed to the Core Top Limit. The IN-CORE light is ON.

A reactor scram due to low RPV water level occurs. One (1) minute later, an operator observes:

- TIP valve indication on Containment Isolation display (Panel 9-3) is RED
- IN-SHIELD light for TIP "A" is ON at Panel 9-13
- Drywell pressure is normal

What action is ***required***?

a. Fire TIP "A" shear valve.

b. Close TIP "A" ball valve.

c. Manually retract TIP "A" to fire the shear valve.

d. Manually retract TIP "A" to close the ball valve.

Answer 26

b. Close TIP "A" ball valve.

The indications at Panel 9-3 indicate that a ball valve is open. If red light (Panel 9-3) stays on, at least one TIP ball valve has not closed. After automatic withdrawal of the TIP on the PCIS group 2 isolation signal, the ball valve failed to automatically close.   This failed automatic action requires immediate operator action to manually perform the ball valve closure. The procedure directs the operator to attempt to manually retract TIP. Since the TIP is already retracted (IN-SHIELD light is on), this action is not necessary.
If ball valve cannot be closed and there are indications of a reactor coolant leak in drywell (as evidenced by the high drywell pressure) then fire appropriate shear valve by operating appropriate keylock switch.

Per 2.0.1.2, step 2.3 "Operators shall ensure automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals per their safety function design.

Upon recognition of a failure of automatic safety feature, Operators shall manually perform those actions necessary to fulfill the safety function."

REFERENCE: 4.1.4; Section 6 and Attach. 1, step 2.1.6, 2.0.1.2 step 2.3

Foils:

a. There is no indications of a LOCA and no attempt has yet been made to close the ball valve.
c. The TIP has already retracted and retracting the TIP does not fire the shear valve.
d. The TIP has already retracted.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0023102001110A Describe the TIP system design features and/or interlocks that provide for the following: Primary containment isolation

COR0023102001130C Given a TIP system control manipulation, predict and explain the changes in the following parameters: Valve status

COR0023102001140H Predict the consequences of the following on the TIP system: High primary containment pressure

COR0023102001160B Given plant conditions, determine if any of the following TIP actions should occur: Ball valve closure

Skills

215001.K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7) Primary containment isolation system: Mark-I&II (Not- BWR1) (3.1/3.4)

Question 27   1068   (1 point(s))
Given the following conditions:

- A Reactor Startup is in progress.
- Power is rising with a stable period.
- SRM detectors are withdrawn except for SRM "A" which fails to withdraw.
- The SRM UPSCALE OR INOPERATIVE alarm has been received.
- SRM "A" is **NOT** bypassed.

If power continues to rise, what is the FIRST point that control rods will be able to be withdrawn?

a. ALL IRMs are on Range 2.

b. ALL IRMs are on Range 3.

c. ALL IRMs are on Range 8.

d. The Mode switch is placed to RUN.

Answer 27

c. ALL IRMs are on Range 8.

IRMs on range 8 will bypass the SRM Rod Block.

Reference: COR0023002, 4.1.1

Foils:

a.b. SRM Rod Block is not bypased
d. Question asked for the FIRST point the control rods can be moved.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0023002001060F Describe the SRM system design features and/or interlocks that provide for the following: IRM/SRM interlock

Skills

215004.A3.04 Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status (3.6/3.6)

Question 28   1080   (1 point(s))

Given the following conditions:

- A plant startup is in progress.
- The Recirculation flow input signal to the Average Power Range Monitors (APRMs) is 50%.
- As Recirculation flow is raised, the output signal from the "B" Flow Unit remains at 50%.
- Actual Recirculation loop flows respond as expected.

As Recirculation flow continues to be raised, what will be the FIRST effect on plant operation from the APRMs/Flow Unit?

a. A half scram will occur due to flow biased neutron flux high.

b. A control rod block will occur due to a Flow Unit Comparator trip.

c. A control rod block will occur due to flow biased neutron flux high.

d. A half scram will occur due to a Flow Unit UPSCALE OR INOP trip.

Answer 28

b. A control rod block will occur due to a Flow Unit Comparator trip.

EXPLANATION OF ANSWER: A Flow Unit Comparator trip will occur when the A Flow Unit exceeds 60% (10% difference)

REFERENCES: STCOR002-01-02, page 16, section III.B, rev. 12.
2.3_9-5-1 9-5-1/A-4

Foils:

a. The margin to the SCRAM is large such that the Comparator trip will occur first.
c. The margin to the APRM rod block is large such that the Comparator trip will occur first.
d. Flow units do not directly cause scram signals.

Bank

Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0020102001050G   Describe the interrelationships between the Average Power Range Monitor System and the following: Flow converter/comparator network

Skills

215005.K1.16   Knowledge of the physical connections and/or cause- effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Flow converter/comparator network: Plant-Specific (3.3/3.4

Question 29 1120 (1 point(s))

Given the following conditions:

- The plant is performing a startup following a refueling outage
- The Reactor Mode Switch is in START & HOT STBY
- The "A" Reactor Protection System (RPS) MG set is being returned to service following maintenance
- Power is on IRM Range 3.
- When the operator placed the RPS Bus "A" Power Supply Selector to "NORMAL," a full reactor scram resulted (all rods in)

What additional condition would have caused this full reactor scram?

a. The Reactor Protection System shorting link switches are closed.

b. IRM Channel "E" was upscale and **NOT** bypassed during the transfer.

c. A Division 2 PCIS Group 1 half isolation was present during the transfer.

d. APRM Channel "F" was upscale and **NOT** bypassed during the transfer.

Answer 29

d. APRM Channel "F" was upscale and **NOT** bypassed during the transfer.

REFERENCES: STCOR002-01-02
PR 4.1.3, page 2, section 2.2.3, rev. 18.

Foils:

a. The shorting link switches are normally closed.
b. Channel E is fed by 24 VDC "A".
c. RMS in "Startup/Hot Standby" bypasses MSIV closure scram.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0020102001060A State the electrical power supplies to the following: LPRM channels

COR0020102001060B State the electrical power supplies to the following: APRM channels

Skills

215005.K2.02 Knowledge of electrical power supplies to the following: (CFR: 41.7) APRM channels (2.6/2.8)

Question 30   16433   (1 point(s))

A power ascension is in progress. Annunciator 9-5-1/D-4 RBM UPSCALE/INOP alarms.
At Panel 9-14, using the Meter Function Switch on RBM "A", the following indications are observed:

- AVERAGE 89%
- APRM SIG 78%
- COUNT 2.0 v (25 % of assigned inputs)
- BLOCK 94%

What is the status of RBM "A"?

a. RBM is inoperable from too few LPRM inputs.

b. RBM is operable, but an edge rod has been selected.

c. RBM is inoperable from a failure to null with the APRM.

d. RBM is operable, but local power is above the APRM signal.

Answer 30

a. RBM is inoperable from too few LPRM inputs.

Too few LPRM inputs count circuit shows less than 50%

REFERENCE: 2.3_9-5-1, 4.1.5; 1.2.4.1

Foils:

b. RBM is inoperable and edge rod would not cause the annunciator.
c. A failure to null has not occurred, there are too few LPRM inputs.
d. RBM is inoperable, local power is below APRM power but this is not the cause for a block, the RBM power is below the block setpoint.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0022402001130A   Given plant conditions, determine if any of the following should occur: RBM control rod withdrawal block.

COR0022402001130C   Given plant conditions, determine if any of the following should occur: RBM inop trip.

Skills

215002.A3.02   Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Meters and recorders: BWR-3,4,5 (3.1/3.0)

Question 31   16511   (1 point(s))

While operating steady state at 100% power the following indications are observed:

- Reactor power lowers
- Narrow Range reactor water level rises
- Indicated core plate d/p lowers
- Indicated TOTAL core flow rises
- "A" and "B" recirculation loop flows rise by the same amount

Which one of the following failures caused the above conditions?

a. A shroud support access hole cover has failed.

b. One (1) of the Jet pumps has a blocked throat.

c. One (1) recirculation pump's speed has raised to maximum.

d. Flow through a control cell (four fuel bundles) has been blocked.

Answer 31

a. A shroud support access hole cover has failed.

A shroud support access hole cover is ~ 19" in diameter. The effect to recirculation flow and core power by the separation of a cover could be significant. If a cover should separate, a flow path which bypasses the core is established which reduces the hydraulic resistance to flow through the core. This condition would indicate an increase in total core flow but actual flow through the core would drop and cause power to drop.

REFERENCES: 2.4RXPWR; 5.3, 6.2, 6.3

Foils:

b. Loop flows will lower in one loop and reactor water level change would not be discernible.
c. Would not provide these indications.
d. This would lower core flow.

Bank
Difficulty 3

Cognitive Level 1

Enabling Objectives

COR0011502001060C   Given plant operating status, predict and explain the changes in the following parameters associated with the following: Reactor Power

COR0011502001060A   Given a specific Nuclear Boiler system malfunction, determine the effect on any of the following: Reactor Water Level

COR0021502001020I   Describe the interrelationships between NBI and the following: Reactor Vessel

Skills

290002.K3.07   Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: (CFR: 41.7 / 45.4) Nuclear boiler instrumentation (3.1/3.1)

Question 32   17860   (1 point(s))

The plant is operating at 50% power when a DEH malfunction causes the Main Turbine Bypass valves to rapidly open.

How is Wide Range RPV water level indication affected and why?

The __________ in flow resistance in the core region causes a/an _________ in the mass above the variable leg tap, causing a ________ in indicated RPV water level.

a. increase, increase, rise

b. increase, decrease, reduction

c. decrease, increase, rise

d. decrease, decrease, reduction

Answer 32

a. increase, increase, rise

Reference: COR002-15-02, 4.6.1, Attachment 2.

Foils:

b. The mass above the variable leg increases and indicated level rises.
c. The flow restriction rises.
d. The flow restriction rises, the mass above the variable leg increases and indicated level rises.

New
Difficulty 4
Cognitive Level 3

Enabling Objectives

COR0021502001040K Briefly describe the following concepts as they apply to NBI: Effects on level indication due to rapid changes in void fraction

Skills

216000.K5.12 Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.5 / 45.3) Effects on level indication due to rapid changes in void fraction (3.2/3.3)

Question 33   16435   (1 point(s))

The plant is operating at power with the following reactor vessel level control alignment:

- RFC-LC-83, MASTER LEVEL CONTROLLER in balance
- RFC-MA-84A, FW CONTROLLER STATION A in balance
- RFC-MA-84B, FW CONTROLLER STATION B in balance
- Feedwater flow is approximately $9.6x10^6$ lbm/hr.
- Steam flow is approximately $9.6x10^6$ lbm/hr.
- RPV water level is +35 inches.
- ALL listed controllers have been nulled.

The Master Controller OUTPUT slowly fails downscale. The operator places the "A" and "B" RFP controller mode switches in MANUAL when RPV water level lowers to +27 inches.

Assuming NO additional action is taken by the operator, what is the response of Feedwater Flow and RPV water level?

Feedwater flow . . .

a. rises to $9.6x10^6$ lbm/hr. Level rises to +42 inches.

b. rises to $9.6x10^6$ lbm/hr. Level remains at +27 inches.

c. rises above $9.6x10^6$ lbm/hr. Level rises to +35 inches.

d. remains below $9.6x10^6$ lbm/hr. Level continues to lower.

Answer 33

b. rises to $9.6x10^6$ lbm/hr. Level remains at +27 inches.

REFERENCE: 2.2.28; Attachment 1, Sections 1.2.5, 1.2.6 and 2.4RXLVL

Foils:
a. Level will not rise.
c. Feed flow will not rise above $9.6x10^6$ lbm/hr. Level will not rise. No recirc runback occurs.
d. Feed flow rises to $9.6x10^6$ lbm/hr. Level does not lower.

Bank
Difficulty 3
Cognitive Level 3

Enabling Objectives

COR0023202001070A Given a RVLC system control manipulation, predict and explain the changes in the following parameters: RPV water level

COR0023202001070B Given a RVLC system control manipulation, predict and explain the changes in the following parameters: RFP Speed/Feed Flow

Skills

259002.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water level (3.8/3.8)

Question 34   1644   (1 point(s))

What effect (if any) will a high intake radiation have on the Main Control Room HVAC system?

a.   There is no automatic response to this signal.

b.   The system suction isolates and air is recirculated through the normal flowpath.

c.   The system lineup shifts to process all outside air through the emergency filter train.

d.   The system suction isolates and air is recirculated through the emergency filter train.

Answer 34

a.   There is no automatic response to this signal.

(System mod removed intake monitor from initiation logic and replace it with Group 6 signal)

REFERENCES: COR0010802

Foils:

b.   The system does not respond to high intake radiation anymore.
c.   The system does not respond to high intake radiation anymore.
d.   The system does not respond to high intake radiation anymore.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

COR0010802001120A   Describe the control Room HVAC design features and interlocks that provide for the following: Control room HVAC reconfigurations

COR0010802001140A   Briefly describe the following concepts as they apply to Control Room HVAC: Airborne contamination (e.g., radiological,toxic gas, smoke) control

COR0010802001200A Predict the consequences of the following items on the Control Room HVAC: Initiation/reconfiguration

Skills

295038.EA1.07 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) Control room ventilation: Plant-Specific. (3.6/3.8)

Question 35 16512 (1 point(s))

The plant is at 100% power. Reactor Building Ventilation is operating as follows:

- EF-1-RB, Exhaust Fan, running
- SF-R-1A-B, Supply Fan, running
- BF-R-1B, Exhaust Booster Fan, running
- ALL other fans are selected to STBY
- HV-DPIC-835A is in AUTO with reactor building pressure is -0.30" wg

How will the Reactor Building Ventilation system respond if the reactor building pressure degrades to -0.25" wg?

a. Supply Fan SF-R-1A-A starts and the supply fan vortex dampers position to correct the reactor building pressure.

b. Exhaust Fan EF-1-RA starts and the exhaust fan vortex dampers position to correct the reactor building pressure.

c. Operating fan lineup does NOT change. Supply fan vortex dampers position to correct the reactor building pressure.

d. Operating fan lineup does NOT change. Exhaust fan vortex dampers position to correct the reactor building pressure.

Answer 35

d. Operating fan lineup does NOT change. Exhaust fan vortex dampers position to correct the reactor building pressure.

Exhaust fan, exhaust booster fan, and supply fan will auto start when the control switch is in standby and Reactor Building pressure is below -0.15" wg and above -0.35" wg. This threshold has not been reached. The exhaust fan vortex dampers will position to improve the reactor building pressure.

REFERENCE: 2.2.47

Foils:

a. No fans auto start. Exhaust dampers reposition.

b. No fans auto start.
c. Exhaust dampers reposition.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0010802001110C Describe the HVAC design features and interlocks that provide for the following: Automatic starting and stopping of fans

COR0010802001130B Describe the control Room HVAC design features and interlocks that provide for the following: Differential pressure control

COR0010802001140B Briefly describe the following concepts as they apply to Control Room HVAC: Differential pressure control

Skills

295035.EA1.01 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.7 / 45.6) Secondary containment ventilation system (3.6/3.6)

Question 36   16496   (1 point(s))

The plant is at 100% power with the following conditions:

- Standby Gas Treatment (SGT) Exhaust Train A is being placed in service to support a surveillance test
- The control switch for SGT Fan 1F is in STBY
- The control switch for SGT Fan 1E is placed in RUN

If SGT-AO-251 (SGT Train A Outlet Valve) remains closed, how will SGT train B respond over the next one (1) minute?

"B" Standby Gas Treatment fan 1F . . .

a. remains off. SGT-AO-252 (SGT Train B Outlet Valve) opens.

b. starts on low flow. SGT-AO-252 (SGT Train B Outlet Valve) opens.

c. remains off. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

d. starts on low flow. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

Answer 36

c. remains off. SGT-AO-252 (SGT Train B Outlet Valve) remains closed.

Low flow in a train will cause the standby fan to start if it is in STBY, the operating train flow is <800 scfm, and a group 6 isolation signal is present or sealed in. There is no group 6 isolation signal for the conditions presented. No conditions will develop on the operating train to cause it to trip within 1 minute.

REFERENCE: 2.2.73

Foils:

a. SGT-AO-252 (SGT Train B Outlet Valve) does not open.
b. SGT B remains off.
d. SGT B remains off.

Modified
Difficulty 2
Coginitive Level 2

Enabling Objectives

COR0022802001080A Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

COR0022802001090B Briefly describe the following concepts as they apply to the Standby Gas Treatment system: Air operated valves operations

Skills

261000.K4.01 Knowledge of STANDBY GAS TREATMENT SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Automatic system initiation (3.7/3.8)

Question 37   1494   (1 point(s))

The turbine building HVAC was operating normally with all fan control switches in AUTO. A sudden atmospheric disturbance caused turbine building pressure to go 1" positive. The HVAC dP controller returned pressure to normal in 25 seconds.

What is the status of the turbine building supply and exhaust fans?

All supply fans . . .

a.   **AND** exhaust fans are tripped.

b.   **AND** exhaust fans remain running.

c.   are tripped, all exhaust fans remain running.

d.   remain running, all exhaust fans are tripped.

Answer 37

b.   **AND** exhaust fans remain running.

REFERENCE: STCOR001-08-01, page 24, section II.F.3, Proc. 2.2.49, step 2.3 & 2.9

Foils:

a.   45 second delay on trip for excessive dp
c.   45 second delay on trip for excessive dp
d.   45 second delay on trip for excessive dp

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

COR0010802001220F   Given plant conditions, determine if the following should occur: Turbine Building Supply Fan trip

Skills

288000.A3.01 Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: (CFR: 41.7 / 45.7) Isolation/initiation signals (3.8/3.8)

Question 38   17901   (1 point(s))

With the plant at full power, the following Reactor Building vent exhaust plenum radiation monitor readings exist:

- RMP-RM-452A: 14 mrem/hr
- RMP-RM-452B: 7 mrem/hr
- RMP-RM-452C: 11 mrem/hr
- RMP-RM-452D: 13 mrem/hr

NO group isolations or automatic initiations occur.

What actions are required (if any) and why?
(Note: Use *actual* setpoints in your evaluation.)

a. **NO** actions are required because **neither** ***DIVISION*** logic has actuated.

b. **NO** actions are required because **only** the ***DIVISION I*** logic has actuated.

c. Manually start **only** "A" SGT train because **only** the ***DIVISION I*** logic has actuated.

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

Answer 38

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

Justification: If RMP-RM-452A or C AND RMP-RM-452B or D exceed 10 mrem/hr, Reactor Building isolates, and both SGT systems start. Per 2.0.3 "Operators shall validate automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals"

REFERENCE: 2.1.22, 2.2.73; 1.3.1.2 (logic), 2.3_9-4-1; Set Points

Foils:

a,b,c Both Divisions should have actuated. The reactor building should have isolated and both SGT trains should have started.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

COR0022802001080A Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

COR0022802001130A Given plant conditions, determine if any of the following should occur: SGT automatic initiation

COR0011802001100D Given a control manipulation, predict and explain the changes to the following Radiation Monitoring systems: Reactor Building Vent Exhaust Plenum radiation monitoring system

COR0011802001120C Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: PCIS Group 6 Isolation

COR0011802001120D Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Standby Gas Treatment Startup

COR0011802001120E Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Reactor Building Ventilation Isolation

Skills

290001.A2.03 Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal...: (CFR: 41.5 / 45.6) High area radiation (3.4/3.6

Question 39   1112   (1 point(s))

DG-1 is paralleled to 4160V Bus 1F for testing. The operator is in the process of adjusting load and voltage when the Governor Control switch sticks in the LOWER position.

If NO operator action is taken, what will be the Diesel Generator response to this condition?

DG output frequency will . . .

a. lower **AND** then the diesel will trip on overcurrent.

b. lower **AND** then the diesel will trip on reverse power.

c. remain constant **BUT** the diesel will trip on overcurrent.

d. remain constant **BUT** the diesel will trip on reverse power.

Answer 39

d. remain constant **BUT** the diesel will trip on reverse power.

EXPLANATION OF ANSWER: Due to the diesel being paralleled the frequency will remain constant but load will be removed causing a reverse power trip which will trip the diesel generator lockout.
a,b. Frequency will remain constant. c. Lowering on the governor control will remove load not increase load.

REFERENCES: STCOR002-08-02, page 34, section III.B.1, IV.4.C, rev. 10.
PR 2.3.2.8, page 4, section C-1/A-4, rev. 20.
PR 2.2.20, page 13, 8, section 4.4, 2.4.3.6, rev. 45.

Foils:

a. frequency will not lower and diesel will not trip on overcurrent.
b. frequency will not lower.
c. diesel will not trip on overcurrent.

Bank
Difficulty 2
Coginitive Level 3

Enabling Objectives

COR0020802001090A   Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Diesel Generator Trips (Normal)

COR0020802001090C   Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Speed Droop Control

COR0020802001130A   Predict the consequences of the following items on the Diesel Generator: Parallel Operation of Diesel Generator

Skills

264000.K3.02   Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: (CFR: 41.7 / 45.4) A.C. electrical distribution (3.9/4.0)

Question 40   16439   (1 point(s))

The plant is at 100% power near the end of cycle with all control rods fully withdrawn.
When performing an APRM functional test (1/2 scram), the scram inlet valve (CRD-AOV-126) for control rod 30-31 opens.

What is the effect (if any) on reactor power and the Scram Discharge Volume (SDV) over the next five (5) minutes and why?

a. The reactor will scram due to high Scram Discharge Volume level.

b. Reactor power and SDV level are unaffected because **NO** control rod motion occurs.

c. The control rod inserts causing reactor power to lower, but power operation continues as SDV level is unaffected.

d. The control rod inserts causing reactor power to lower, but power operation continues and SDV level rises but remains below the scram setpoint.

Answer 40

c. The control rod inserts causing reactor power to lower, but power operation continues as SDV level is unaffected.

Control rod inserts. SDV level is unaffected because the over piston area volume
is evacuated to the reactor vessel past the CRDM seals.

REFERENCE:
2.4CRD direct troubleshooting per 2.2.8.
2.2.8; section 25, Attachment 2 - 1.2.4.2

Foils:

a. A reactor scram will not occur. The SDV level will not change.
b. The control rod will insert into the core. A single control rod inserting will reduce reactor power.
d. No leakage will occur into the SDV.

Bank
Difficulty 3

Cognitive Level 2

Enabling Objectives

COR0020402001110H Predict the consequences a malfunction of the following would have on the CRDH systems: Leaking scram valves.

Skills

201003.A1.01 Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: (CFR: 41.5 / 45.5) Reactor power (3.7/3.8)

Question 41   16414   (1 point(s))

The plant is operating at 80% power. The CAM alarms in the Off-Gas Building and the following alarms are received:

- B-3/A-3, CONDENSER AIR REMOVAL ISOLATION
- B-3/B-3, CONDENSER AIR REMOVAL HIGH TEMP
- B-3/B-4, CONDENSER AIR REMOVAL HIGH PRESSURE
- K-1/A-4, OFFGAS FILTER HIGH D/P

What event has occurred?

a. High Off-Gas activity.

b. Fire in the Off-Gas Building.

c. Explosion in the Off-Gas System.

d. Trip of the Off-Gas Dilution Flow Fans.

Answer 41

c. Explosion in the Off-Gas System.

Justification: These alarms are consistent with an explosion in the off gas system and the CAM alarm for the hi radiation in the off gas building.

REFERENCE: 2.4OG 5.0; 2.3_B-3/A-3/B-3/B-4 section 3.2

Foils:

a. This would cause the isolation but not the high temperature and pressure.
b. This would not cause the isolation or the high pressure or the CAM alarm.
d. This would not cause the isolation or high temperature or the CAM alarm.

Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

COR0011602001100G   Predict the consequences of the following items on the Off Gas system: Condenser Air Removal failure

SKL012411600A030D   Given plant conditions, predict changes in the following Off-Gas system components/parameters: System flows.

SKL012411600A030E   Given plant conditions, predict changes in the following Off-Gas system components/parameters: Process radia tion monitoring indications.

Skills

271000.A3.05   Ability to monitor automatic operations of the OFFGAS SYSTEM including: (CFR: 41.7 / 45.7) System indicating lights and alarms (2.9/2.9)

Question 42   17884   (1 point(s))

The station was in a startup at 57% power when main condenser vacuum rapidly degraded. When vacuum degraded to 22.5" Hg, the CRS directed a manual reactor scram be inserted and then the main turbine be tripped.

Why was the reactor scrammed ***before*** the main turbine is tripped under these conditions?

To prevent . . .

a.	forcing an automatic protective action.

b.	a rapid depressurization of a critical reactor due to bypass valves opening fully.

c.	the need to close the MSIVs which would add heat to the primary containment.

d.	reducing MCPR below the operating limit due to the reduction in feedwater heating.

Answer 42

a.	forcing an automatic protective action.

Reference 2.4VAC, Operations strategy (2.0.3, 2.0.1.2)

Foils:

b.	this is not a basis for the scram.
c.	The MSIVs still may need to be closed.
d.	MCPR is not the basis for the scram.

NEW
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT0320132K0K0100 Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

INT0320132J0J0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

295002.AK3.01 Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM: (CFR: 41.5 / 45.6) Reactor SCRAM: Plant-Specific (3.7/3.8)

Question 43   17888   (1 point(s))

The plant was operating at 100% power when 4KV bus 1F de-energized. The reactor is scrammed per the alarm card. After the immediate scram actions were complete, the Startup Transformer tripped. The following conditions exist:

- Groups 2, 3 and 6 isolations occurred
- HPCI and RCIC did not automatically start
- 87 control rods are between notch 12 and notch 36
- APRM downscales are in

What action(s) (if any) are required and why?

a. Enter EOP 1A, "RPV Control". Do **NOT** enter 5.3EMPWR "Emergency Power" or 2.4CRD "CRD Trouble." Applicable Abnormal/Emergency procedures are **NOT** entered if EOP entry **IS** required.

b. Enter 5.3EMPWR "Emergency Power", 2.4CRD "CRD Trouble" **AND** EOP 1A, "RPV Control." Do **NOT** execute any Abnormal procedure steps if they conflict with any steps in the EOPs.

c. Enter 5.3EMPWR "Emergency Power" and 2.4CRD "CRD Trouble". Do **NOT** enter EOP 1A, "RPV Control" as EOP entry is **NOT** required if the specific condition is adequately addressed by the applicable Abnormal/Emergency procedure.

d. Enter EOP 1A, "RPV Control", 5.3EMPWR "Emergency Power" and 2.4CRD "CRD Trouble". EOPs and applicable Abnormal/Emergency procedures must be entered and ALL Abnormal and EOP procedure steps must be completed.

Answer 43

b. Enter 5.3EMPWR "Emergency Power", 2.4CRD "CRD Trouble" **AND** EOP 1A, "RPV Control." Do **NOT** execute any Abnormal procedure steps if they conflict with any steps in the EOPs.

REFERENCE: 2.0.1.2 step 2.7. 0.1, 5.8 step 3.2.1, 3.2.6

Foils:

a. Applicable Abnormal/Emergency procedures must be entered.
c. Applicable EOPs must be entered
d. ALL the AOP and EOP steps do NOT have to be performed if they are in conflict the EOPS take precedent.

K/A 295037 (SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown) 2.4.8
New
Cognitive Level: 2
Difficulty: 2

Enabling Objectives

INT032010100E0300 Describe the heirarchy between the Emergency Operating Procedures, Abnormal Procedures, and Emergency Procedures, including which guidance takes precedence.

Skills

2.4.8 Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs. (CFR: 41.10 / 43.5 / 45.13)

Question 44   4243   (1 point(s))

A weekly surveillance test was completed at 1300 on Tuesday of last week.

In accordance with procedure 0.26 (Surveillance Program), what is the **LATEST** this test can be scheduled to be completed this week, without exceeding an LCO?

a.   1300 on Wednesday

b.   0100 on Thursday

c.   0700 on Thursday

d.   1300 on Thursday

Answer 44

c.   0700 on Thursday

A deviaiton of 25% of the weekly surveillance interval is allowed (42 hrs or 1.75 days)

Foils:

a.   24 hrs
b.   36 hrs
d.   48 hrs

REFERENCE: PR 0.26 Surveillanve Program Page 10 Section 8.1.5

Bank
Difficulty 2
Cognitive Level: 1

Enabling Objectives

INT032010100G010J   Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test performance

INT032010100G010M   Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Frequency

INT032010100G010L Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Precautions and Limitations

Skills

2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Question 45   5131   (1 point(s))

Which one of the following activities can have its concurrent/independent verification waived?

a. A local control switch must be placed in Auto in a contaminated area as part of a tagout restoration.

b. A 4160 VAC breaker already verified in test will have a test block and extention arm installed.

c. A 480 VAC breaker for non-safety equipment will be racked out for a tagout requiring 25 mrem exposure to the performer.

d. A valve must verified open, requiring 12 mrem exposure to the performer when performing a periodic system component checklist.

Answer 45

d. A valve must verified open, requiring 12 mrem exposure to the performer when performing a periodic system component checklist.

Component checklist item with a dose exposure in excess of 10 mrem.

Reference: 0.31

Foils:

a. Required by procedure.
b. Required by procedure.
c. Not an exception.

Bank
Difficulty 3
Cognitive Level 1

Enabling Objectives

SKL00803020010200   Describe the three forms of verification used to satisfy the requirements of the equipment status control program and when each form of verification is required or permissible.

INT032010100H010C   Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: Independent/Concurrent verification

INT032010100H010L Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: Breaker position verification

Skills

2.1.29 Knowledge of how to conduct and verify valve lineups. (CFR: 41.10 / 45.1 / 45.12)

Question 46   16431   (1 point(s))

Following seven (7) days of vacation, an operator works twelve (12) hours on the first day back on shift and then works an additional four (4) hours of overtime.

Which one of the following describes the MAXIMUM number of hours this operator can work the next day WITHOUT exceeding the CNS working hour limitations?

a.	4 hours.

b.	8 hours.

c.	12 hours.

d.	16 hours.

Answer 46

b.	8 hours.

The restrictions are 16 hours in any 24-hour period and 24 hours in any 48-hour period. Since the operator worked 16 hours on the first day, then without any extension the operator is restricted to 8 hours on the second day of work.

REFERENCE: 0.12, 4.2.2

Foils:

a.	Can work 8 hours.
c.	Can only work 8 hours without an approved extension.
d.	Can only work 8 hours without an approved extension.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT032010100F0100	State the working hours limitations and approval requirements associated with Administrative Procedure 0.12, Working Hours Limitations.

INT032010100F0200 Given the previous working hours/days history of an individual, determine if the individual is in compliance with the working hours limitations set forth in Administrative Procedure 0.12, Working Hours Limitations.

Skills

2.1.1 Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

Question 47   5654   (1 point(s))

A LOCA has occurred and the following conditions exist:

- RPV level is -120" and stable on the wide range instruments.
- Attempts at level restoration are in progress, but currently there are no means of injection available.
- Annunciator 9-3-1/A-1 ADS TIMERS ACTUATED is energized.
- RPV pressure is 490 psig and lowering.

What is the basis for inhibiting ADS at this time?

a. To prevent large amounts of uncontrolled injection into the RPV.

b. The ADS logic should not have actuated under current plant conditions.

c. The operator has more information as to when the RPV should be depressurized.

d. To allow the crew to rapidly depressurize the RPV to the main condenser and prevent adding significant heat to the suppression pool.

Answer 47

c. The operator has more information as to when the RPV should be depressurized.

REFERENCE: INT0080609 EOP basis, PR 2.3.2.22

Foils:

a. This is a basis for terminating and preventing injection during an ATWS.
b. These RPV pressure conditions would require ADS.
d. The existance of adequate core cooling at the time of ADS initiation is not a basis for inhibiting ADS.

Modified
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00806090011200 Given an EOP flowchart 1A, RPV CONTROL step, state the reason for the actions contained in the step.

Skills

2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 45.13)

Question 48   5349   (1 point(s))

A Loss of Coolant Accident has occurred with the following conditions:

-	Reactor pressure	470 psig (lowering slowly)
-	Indicated Wide Range Reactor water level	-120" (steady)
-	Drywell pressure	5.5 psig (rising slowly)
-	Drywell temperature	350° F (all points) (steady)

What is the status of Wide Range Reactor Level Instrumentation?

Wide Range Reactor Level Instrumentation is . . .

a. accurate **AND** can be used for trending.
Actual Reactor level is -120".

b. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is higher than -120".

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is lower than -120".

d. **NOT** accurate **AND CANNOT** be used for trending.

Answer 48

c. **NOT** accurate **BUT** can be used for trending.
Actual Reactor level is lower than -120".

EXPLANATION OF ANSWER: c. is correct. Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes. The elevated Drywell temperatures and increased density of RPV water combine to cause indicated level to be erroneously high and actual Reactor level will be lower than -120".

REFERENCES: EOP Graphs 1 & 15, CAUTION 1
INT0080618 INT0080605

Foils:

a. Indicated level is erroneously high and actual Reactor level will be lower than 120".
b. Actual Reactor level will be lower than -120".
d. Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes.

Provide to Candidate: EOP Graphs 15A, B, C, D & E

Bank
Difficulty 4
Cognitive Level 3

Enabling Objectives

INT00806050011000 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

INT00806180010200 For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

INT00806050011200 Given plant conditions, assess if RPV water level can be determined or not.

Skills

295028.EK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: CFR: 41.8 to 41.10) Reactor water level measurement (3.5/3.7)

Question 49   5268   (1 point(s))

Given the following conditions:

- A small steam leak has occurred in the Drywell.
- Drywell temperature is 165° F and rising.
- Drywell pressure is 2.6 psig and rising.
- EOP-3A, Primary Containment Control has been entered.

What action is required to reduce Drywell temperature for these conditions?

a. Vent the Drywell.

b. Initiate Drywell sprays.

c. Emergency Depressurize the RPV.

d. Operate all available Drywell Cooling.

Answer 49

d. Operate all available Drywell Cooling.

EXPLANATION OF ANSWER: This is the first action specified by the Drywell Temperature leg of EOP-3A.

REFERENCE: EOP-3A

Foils:

a. With a LOCA inside containment, venting the Drywell is not allowed.
b. Drywell Sprays are not permitted by the Drywell Spray Initiation Limit Graph.
c. Emergency Depressurization is not yet required.

Bank
Difficulty 2
Cognitive Level 2

Provide to Candidate: EOP 3A with entry conditions and CAUTIONS Removed.

Enabling Objectives

INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

INT00806130101000 Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Skills

295012.AK2.02 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8) Drywell cooling. (3.6/3.7)

Question 50   17856   (1 point(s))

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- RCIC cannot be isolated
- RCIC temperatures in the NE Quad are 214°F and rising
- RCIC Room radiation levels are 300 mr/hr and rising
- A reactor scram is inserted and all control rods fully insert

A high reactor building ventilation exhaust radiation level had isolated reactor building ventilation, but the condition has cleared and annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD reset.

What is the required EOP response to this annunciator clearing and the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Ensure Reactor Bldg. HVAC remains isolated because EOP 1A requires a group 6 isolation.

d. Ensure Reactor Bldg. HVAC remains isolated until RP ensures normal radiation levels to minimize the spread of contamination.

Answer 50

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

REFERENCE: EOP 5A, PSTG/SATG, 2.3_9-4-1

Foils:

a. This is <u>not</u> a bases for restarting Reactor Bldg. HVAC and Rx Bldg ventilation releases are NOT considered elevated.
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

Bank
Difficulty 3
Cognitive Level 2

Provide to Candidate: EOP 5A with entry conditions and Cautions removed.

Enabling Objectives

INT00806170010600 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

INT00806170010700 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Skills

295034.EK2.04 Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following: (CFR: 41.7 / 45.8) Secondary containment ventilation. (3.9/3.9)

Question 51   17157   (1 point(s))

The plant is operating at 100% power with "B" Narrow Range level indicator selected for level control. The following conditions occur:

- "A" and "C" Narrow range are 5" and lowering
- "B" Narrow range is 40" and rising
- Steam Nozzle is downscale

30 seconds later,

- "A" & "C" Narrow range are downscale
- "B" Narrow range is upscale
- "A" wide range is -10"
- "B" wide range is upscale
- ALL -42" and -113" alarms are clear
- Reactor Pressure 995 psi
- Drywell Temperature 125 °F

At the 30 second data time, can RPV water level be determined, and if so, what is actual RPV water level?

RPV water level _________ be determined. Actual RPV water level is ____________.

a.	CANNOT, unknown

b.	CAN, above +60"

c.	CAN, -10"

d.	CAN, below -42" and above -113"

Answer 51

c.	CAN, -10"

The conditions provided are indicative of a "B" side reference leg failure. There are multiple indications that RPV water level is above TAF.

REFERENCE: INT0080605, Attachment 2

Foils:

a. RPV water level CAN be determined
b. RPV water level is -10"
d. RPV water level is -10"

Bank
Difficulty 4
Cognitive Level 3

Enabling Objectives

INT00806090011300 Given plant conditions, assess if RPV water level can be determined or not.

INT00806050011200 Given plant conditions, assess if RPV water level can be determined or not.

Skills

295009.AA2.01 Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Reactor water level. (4.2/4.2)

Question 52   5920   (1 point(s))

Which of the following can be used to verify that the reactor will remain shut down under all conditions without boron ***without*** consulting the Reactor Engineer?

a.   All control rods are inserted to or beyond notch 04.

b.   One (1) control rod is at position 48, all other control rods are full in.

c.   Two (2) control rods are at position 12, all other control rods are full in.

d.   The withdrawn control rods are separated by at least two control rod cells in all directions.

Answer 52

b.   One (1) control rod is at position 48, all other control rods are full in.

Reference: INT0080605

Foils:

a.   Must be inserted to or beyond notch 02.
c.   Shutdown margin may not be met. Not a condition specified to satisfy requirements.
d.   Shutdown margin may not be met. Not a condition specified to satisfy requirements.

Bank
Difficulty: 2
Cognitive level: 1

Enabling Objectives

INT00806050010400   State the criteria used to determine that the reactor will remain shutdown under all conditions without boron injection.

Skills

295015.AK1.01 Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10) Shutdown margin. (3.6*/3.9*)

Question 53   16495   (1 point(s))

During a LOCA concurrent with a Loss of Offsite power, the following conditions exist:

- Drywell Pressure is 60 psig and slowly rising
- Torus Pressure is 58 psig and slowly rising
- PC water Level is 25 feet and stable

Which one of the following methods is required to ***initially*** vent the primary containment for these conditions?

a. Vent the Torus using the hard pipe vent.

b. Vent the Drywell using the 24" ductwork.

c. Vent the Torus using the 1 and 2 inch bypass valves through SGT.

d. Vent the Drywell using the 1 and 2 inch bypass valves through SGT.

Answer 53

c. Vent the Torus using the 1 and 2 inch bypass valves through SGT.

Justification: Per EOP-3A, Primary Containment Control, if Suppression Pool level is < 28.5', then vent the torus using the Torus Vent path and SGT.

Reference: EOP-3A, 5.8.18

Foils:

a. Hard pipe venting is NOT required.
b. The torus should be vented first, use of the 24" ductwork is NOT required.
d. The torus should be vented first.

Bank
Difficulty: 3
Cognitive level: 2

Enabling Objectives

INT00806130010500 Explain why the torus vent path is preferred over the drywell vent path.

INT00806130101000 Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

INT0080613001040A State the basis for primary containment control actions as they apply to the following: Specific setpoints

INT0080613001040B State the basis for primary containment control actions as they apply to the following: Primary Containment Control Systems

INT00806130011200 Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Skills

295010.AA1.05 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6) Drywell/suppression vent and purge (3.1/3.4)

Question 54   16481   (1 point(s))

Given the following conditions:

- The plant has experienced a LOCA with a loss of ALL injection.
- All Control Rods have inserted.
- RPV pressure is being controlled with SRVs.
- RPV water level has lowered to -35 inches (Corrected Fuel Zone).

What is the status of core cooling?

Adequate core cooling exists . . .

a. at this RPV water level.

b. only when the SRVs are closed.

c. only if RPV water level is raised 10 inches.

d. only if injection is established at this water level.

Answer 54

a. at this RPV water level.

At levels above -40 inches with no injection there is sufficient steam flow to provide adequate core cooling.

REFERENCE: EPGS/SATGS, INT008-06-09

Foils:

b. If SRVs close, adequate core cooling is NOT assured with RPV water level below TAF.
c. RPV level does not have to be raised to -25".
d. Just establishing injection will not assure adequate core cooling, level must also be restored to > -25".

Bank
Cognitive Level: 3

Difficulty: 3

Provide to Candidate: EOP 1A with entry conditions and Cautions removed.

Enabling Objectives

INT00806070010800 Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, state the reasons for the actions contained in the steps.

INT00806090010100 Describe the three mechanisms specified in the EOPs to assure adequate core cooling including the RPV water level band required and which is the preferred method.

Skills

295031.EA2.04 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Adequate core cooling (4.6*/4.8*)

Question 55 17900 (1 point(s))

Why is entry into EOPs required if the Reactor Building dP cannot be maintained negative?

This reactor building (RB) dP is an indication that . . .

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

b. the continued operability of equipment needed to carry out EOP actions may be compromised.

c. radioactivity is being released to the environment when the ventilation system should have automatically isolated.

d. an indication that water from a primary system (or from a primary to secondary system leak) may be discharging into the secondary containment.

Answer 55

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

REFERENCE: **INT0080617**

FOILS:
b. This is the basis for the high temperature entry.
c. This is the basis for the high Rx bldg exhaust radiation level.
d. This is the basis for the entry on radiation above Max Normal Operating Level.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00806170010100 List the entry conditions to Flowchart 5A (including the radioactivity release path) and briefly explain each.

Skills

2.3.11 Ability to control radiation releases. (CFR: 45.9 / 45.10)

Question 56   4139   (1 point(s))

What is the basis for stopping and preventing HPCI when suppression pool level lowers to 11 feet, irrespective of adequate core cooling?

a. Prevent HPCI pump damage from low NPSH.

b. Prevent uncovering the downcomers in the torus.

c. Prevent containment pressurization from HPCI exhaust.

d. Prevent loss of the ultimate heat sink by removing too much water.

Answer 56

c. Prevent containment pressurization from HPCI exhaust.

REFERENCE: EOP-3A, INT0080613

Foils:

a. HPCI protected for NPSH with low suction pressure trip.
b. This is the HCLL curve flat area at 9.6'.
d. Other level limits protect this 9.6' for downcomers, 6' for SRVs and HCTL and HCLL.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT00806130010900   Explain why HPCI but not RCIC must be secured at a primary containment water level of 11 feet.

Skills

295030.EK3.02 Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) HPCI operation: Plant-Specific. (3.5/3.7)

Question 57   17896   (1 point(s))

Why is the RPV Emergency Depressurized if Pressure Suppression Pressure (Graph 10) is exceeded?

a. Failure of primary containment may occur if a primary system rupture develops.

b. Failure of primary containment may occur if drywell sprays are initiated.

c. Failure of SRV Tailpipes may occur due to steam bypassing the suppression pool.

d. Failure of SRV Tailpipes may occur due to inadequate differential pressure across the balancing disc.

Answer 57

a. Failure of primary containment may occur if a primary system rupture develops.

REFERENCE: INT0080618, PSTG

Foils:

b. PSP is not based on drywell spray restricitons (DWSIL is).
c. SRVTPLL is not based on steam bypassing the suppression pool.
d. SRVTPLL is not based on dp on balancing disc (PCPL is).

Bank
Difficulty: 3
Cognitive level: 1

Enabling Objectives

INT00806130011200   Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

INT0080613001040C   State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A

INT00806180010200   For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Skills

295024.EK3.04 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.5 / 45.6) ?Emergency depressurization (3.7/4.1)

Question 58   16482   (1 point(s))

Following a control room evacuation due to toxic gas in the control room, operations personnel are at the stations required by the associated procedure.

Who verifies RPV level and RPV pressure?

a. The Control Room Supervisor.

b. The Reactor Building Station Operator.

c. The Control Building/Critical Switchgear Operator.

d. The Turbine Building Switchgear A and B Operators.

Answer 58

a. The Control Room Supervisor.

Justification: Verified at the ASD Panel, which is manned by the SS and CRS.

REFERENCE: 5.1ASD

Foils:

b. Operates SW Valve
c. Checks Critical Buses and isolates RCIC
d. Determines status of Non-critical 4 KV Buses

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

OTH01592020010600 Given a standard set of references available in the Control Room, be able to perform the steps necessary to shutdown the reactor from outside the control room as the ASD room operator.

Skills

295016.AA1.06 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.7 / 45.6) Reactor water level. (4.0/4.1)

Question 59   17858   (1 point(s))

The plant is at 60% power for a rod sequence change.

The RO has just withdrawn control rod 22-19 to notch 24. After the rod control movement switch is released and the SETTLE light turns off, the RO observes control rods 22-19 slowly moving inward. No other control rods are moving.

The following CRD parameters exist:

- CRD Drive Water D/P   350 psid (pegged high)
- CRD Cooling Water flow   60 gpm (pegged high)

What action is required?

a. Scram the reactor.

b. Using EMERGENCY IN, fully insert rod 22-19.

c. Momentarily place the rod movement control switch to WITHDRAW.

d. Using CRD-MO-20, DRIVE PRESSURE CONT VALVE, reduce drive water differential pressure.

Answer 59

b. Using EMERGENCY IN, fully insert rod 22-19.

REFERENCE: 2.4CRD

Foils:

a. The reactor would be scramed if two or more rods were drifting in, not for a single rod.
c. This may temporarily stop the rod movement, but the rod must be driven in.
d. Drive water D/P is high but the action required is to insert the control rod, then diagnose the cause of the rod drift.

Modified
Difficulty 3
Cognitive Level 2

Enabling Objectives

INT0320122I0I0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

INT0320122J0J0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

201001.A2.12 Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those ...: (CFR: 41.5/45.6) High cooling (2.8/2.9)

Question 60   16490   (1 point(s))

The reactor has been shutdown for 18 hours and is currently in Cold Shutdown (MODE 4). A cooldown is in progress with reactor coolant temperature at 162°F. RHR Loop "A" is in Shutdown Cooling with both reactor recirculation pumps tripped. Subsequently, a Group 2 isolation signal occurs and RHR CANNOT be restarted.

Where is RPV water level required to be maintained for the current conditions and why?

a. Above +48 inches on the narrow range RPV level instruments to promote natural circulation.

b. At 0.0 inches on the wide range RPV level instruments to support alternate heat removal using RWCU.

c. Flooded (solid) on the shutdown range RPV level instruments to support alternate heat removal using the SRVs.

d. Between +27.5 inches and +42.5 inches on the narrow range RPV level instruments to minimize thermal stratification in the reactor pressure vessel.

Answer 60

a. Above +48 inches on the narrow range RPV level instruments to promote natural circulation.

REFERENCE:
2.4SDC; 4.9, and attachment 2 (Step 4.4 to step 4.9 to Attachment 2)
2.1.4 precaution 2.9

Foils:

b. Water level is not high enough to support this method of heat removal
c. Not an approved method of heat removal under these conditions.
d. Circulation is needed to minimize thermal stratification.

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT032010400B0400   Discuss cautions and notes associated with Procedure 2.1.4, Normal Shutdown.

INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

295021.AK3.05 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5 / 45.6) Establishing alternate heat removal flow paths (3.6/3.8)

Question 61   16468   (1 point(s))

The unit is operating at 100% power when the following alarms are energized:

- M-1/A-1, REC SYSTEM LOW PRESSURE
- M-1/A-3, REC SURGE TANK LOW LEVEL

Fifty (50) seconds after receipt of the alarms, the appropriate immediate operator actions have been completed. REC pressure is 56 psig and lowering.

What action(s) is/are now required?

a. Trip ALL Drywell FCUs.

b. Isolate REC to the Augmented Radwaste System.

c. Manually scram the reactor and shutdown ALL REC Pumps.

d. Perform rapid power reduction per 2.1.10 to 35 Mlbm/hr core flow.

Answer 61

c. Manually scram the reactor and shutdown ALL REC Pumps.

Justification: These alarms indicate a break in the REC piping that will exceed makeup capacity and result in a loss of ability to cool the recirc pumps and reactor auxiliary equipment. The system must be shutdown, the reactor scrammed, and then limited cooling may be accomplished with one pump.

REFERENCE: 5.2REC   4.1, 4.7 Attachment 2, Step 1.1

Foils:

a. There is no requirement to secure the FCUs.
b. This valve automatically isolates when REC pressure falls to 60.5 psig for 40 seconds which was satisfied based on the alarm condition present for 50 seconds.
d. This would be required by 2.4PC if not for the reactor scram that is required first.

K/A 400000 Component Cooling Water

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT0320126O0O0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

INT0320126Q0Q0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Skills

2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question 62 13338 (1 point(s))

With the plant operating at 100% power, a loss of off site power occurs. Both diesel generators fail to start and CANNOT be started. HPCI and RCIC recover RPV water level to +35" (NR), when flow is stopped on both systems. RPV water level is lowering at an average of 1 inch per minute due to SRV operation.

What procedural restriction (if any) applies to the continued use of HPCI in response to this event until on-site or off-site electrical power can be restored?

a. HPCI suction must be shifted to the suppression pool.

b. HPCI must be placed in RPV Pressure Control Mode.

c. HPCI can be operated until the division 2 battery is exhausted.

d. HPCI must be secured after one cycle of operation and must remain off.

Answer 62

d. HPCI must be secured after one cycle of operation and must remain off.

This procedure assumes that RPV water level and pressure is initially controlled by HPCI, as directed by the EOPs. CNS has committed to secure HPCI after one cycle of operation, even if EOPs ***allow*** HPCI use, in order to extend station battery life during station blackout. HPCI would still be operated if ***required*** by EOPs, but the conditions provided are within RCIC capability.

REFERENCES: 5.3SBO; 1.4 and NOTE, and 5.3

Foils:
a. There is no requirement to shift to the SP
b. HPCI must be secured to save DC power.
c. HPCI must be secured to save DC power.

Modified
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT0060119001150D Describe each of the following special events evaluated in the CNS USAR that could challenge the integrity of the radioactive material barriers: Station Blackout

INT0320131W0W0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

INT00601190012200 Given a specific USAR analyzed Special Event, describe the initial plant condition assumed in the analysis.

INT00601190012500 Given a specific analyzed Anticipated Operational Transient or Special Event, state the appropriate safety actions necessary to prevent exceeding their safety design bases.

Skills

295003.AK1.06 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10) Station blackout: Plant-Specific. (3.8/4.0*)

Question 63   3175   (1 point(s))

Given the following conditions:

- The Plant is operating at 100% power.
- A fire in the Cable Spreading Room has been reported **AND** the Fire Brigade is on the scene.
- The actions of 5.4FIRE, "General Fire Procedure" **AND** 5.4POST-FIRE, "Post-Fire Operational Information" are being performed.
- The Control Room Supervisor enters 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room" as directed by procedure.

Based on these conditions, which choice below describes a condition that will dictate a Turbine Trip be initiated per 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"?

A Turbine Trip should be performed . . .

a. prior to leaving the Control Room, **IF** possible.

b. **ONLY** for a confirmed fire in the Control Room.

c. **ONLY IF** the fire in the Cable Spreading Room is confirmed.

d. **ONLY IF** unexpected DEH/Turbine related actuations occur.

Answer 63

a. prior to leaving the Control Room, **IF** possible.

REFERENCE: PR 5.4FIRE-SD, "Fire Induced Shutdown From Outside Control Room"

Foils:

b. Required prior to leaving if possible, irrespective of fire location.
c. Not a reason to trip the turbine.
d. Not a reason to trip the turbine.

Bank

Difficulty 2
Cognitive Level 1

Enabling Objectives

INT0320134G0G0100 Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

Skills

2.4.27 Knowledge of fire in the plant procedure. (CFR: 41.10 / 43.5 / 45.13)

Question 64   17850   (1 point(s))

A plant shutdown is in progress at 88% power when the following indications are received:

- Steam line flow on the "D" steam line indicates zero (0) flow.
- Annunciator 9-5-2/F-2, REACTOR HIGH PRESSURE alarms.
- Reactor pressure continues to rise.

What action is required?

a. Lower the DEH pressure setpoint.

b. Place Governor valves in MANUAL.

c. Manually scram the reactor at 1030 psig.

d. If only one (1) MSIV is closed, re-open the valve.

Answer 64

c. Manually scram the reactor at 1030 psig.

Manually scram the reactor at 1030 psig per 2.4DEH and 2.3_9-5-2.

Reference 2.4DEH and 2.3_9-5-2.

Foils:

a. This would have no effect and is not in accordance with any procedure.
b. This would not be appropriate and would not correct the problem.
d. An MSIV should not be re-opened without engineering concurrence.

K/A 295025 (High Reactor Pressure), 2.4.11 (3.4)

Bank
Difficulty: 4
Cognitive level: 3

Enabling Objectives

INT0320125L0L0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)

Question 65 10634 (1 point(s))

The plant was operating at 100% power when the following annunciators alarmed:

- A-4/B-5, SERVICE AIR LOW PRESSURE
- A-4/B-4 SERVICE AIR ISOLATION PCV-609
- A-4/A-4, AIR RECEIVER A OR B LOW PRESSURE alarmed

The crew noted INSTRUMENT AIR PRESSURE (IA-PI-606), indicated 75 psig and closed IA-MO-80, NON CRIT INSTRUMENT AIR ISOLATION. Immediately following the closure of IA-MO-80 control room indications are:

- SERVICE AIR PRESSURE (SA-PI-611), indicates 0 psig.
- INSTRUMENT AIR PRESSURE (IA-PI-606), indicates 90 psig and increasing.

Which operator actions(s) is/are required next?

a. Re-open SA-PCV-609 to restore Service Air.

b. Re-open IA-MO-80 to restore non-critical Instrument Air.

c. Scram the reactor and transfer level control to HPCI/RCIC.

d. Manually open "A" Air Compressor TEC Return Isolation, TEC-AOV-21AV.

Answer 65

c. Scram the reactor and transfer level control to HPCI/RCIC.

Even though the closure of IA-MO-80 restored the instrument air pressure the loss of non-critical instrument air supply causes unsafe operation due to the loss of feedwater heating. The reactor should be scrammed as a scram required setpoint has been exceeded, even though air pressure is now above that requiring a scram.

REFERENCE: 5.2AIR; 2.1.5, Attachment 2 2.0.1.2 Step 7.1.4
SER 4-00

Foils:

a. is incorrect. Even though this action could be performed to safely restore service air, a scram is required.

b. is incorrect. Even though this action could be performed to safely restore instrument air, a scram is required.

d. is incorrect. This is a valve that fails closed on a loss of air, but there is no guidance to manually open the valve.

K/A 295019 (Part. Or Comp. Loss of Inst. Air) 2.4.31
Bank
Difficulty 3
Cognitive Level 2

Enabling Objectives

INT0320136N0N0100 Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Skills

2.4.31 Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question 66   7755   (1 point(s))

Which one of the following is the normal, 100% power, lineup of the Containment H_2 and O_2 Monitoring system per 2.2.60.1 and what indications are available?

a. One Division in operation provides **only** O_2 measurement ***capability*** consisting of PMIS displays, O_2 recorders, an O_2 digital indicator.

b. One Division in operation provides H_2 **AND** O_2 measurement ***capability*** consisting of PMIS displays, H_2 **AND** O_2 recorders, an O_2 digital indicator and H_2 annunciation.

c. Two divisions in operation. Both provide H_2 **AND** O_2 monitoring consisting of H_2 **AND** O_2 recorders, a digital O_2 indicator and H_2 annunciation. **NO** input is provided to PMIS.

d. Two divisions in operation. One division provides H_2 monitoring, the other provides O_2 monitoring. An O_2 recorder, digital O_2 indicator and H_2 annunciation are provided. Both divisions provide input into PMIS displays.

Answer 66

b. One Division in operation provides H_2 **AND** O_2 measurement ***capability*** consisting of PMIS displays, H_2 **AND** O_2 recorders, an O_2 digital indicator and H_2 annunciation.

Foils:

a. H2 monitoring is available and annunciated.
c. Only one division in service, PMIS input is provided.
d. Only one division in service monitoring both parameters.

Reference: 2.2.60.1 Discussion Section.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

COR0020302001140D Briefly describe the following concepts as they apply to the Primary containment: Hydrogen/oxygen concentration measurement

Skills

500000.EA1.02 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6) Primary containment oxygen instrumentation (3.3/3.2)

500000.EA1.01 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6) Primary containment hydrogen instrumentation (3.4/3.3)

Question 67   17854   (1 point(s))

The plant is operating at rated power when one (1) inboard MSIV fast closes. What is the expected plant response and why?

Reactor power and pressure will rise due to . . .

a. decreased control rod worths. The reactor will scram.

b. feedback from the void coefficient of reactivity. The reactor will scram.

c. decreased control rod worths. Reactor power will stabilize above 100%, but below the scram setpoint.

d. feedback from the void coefficient of reactivity. Reactor power will stabilize above 100%, but below the scram setpoint.

Answer 67

b. feedback from the void coefficient of reactivity. The reactor will scram.

REFERENCE: Reactor theory, USAR IV-4.

Foils:

a. Reactor pressure will rise because the steam lines have isolated.
c. Reactor power will lower because a scram occurs on an MSIV isolation on low pressure in RUN (<835 psig).
d. Reactor power will lower because a scram occurs on an MSIV isolation on low pressure in RUN (<835 psig)
Reactor pressure will rise because the steam lines have isolated.

New
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT00601190010100 Given a set of initial operating conditions, select those conditions that would tend to make the consequences of an analyzed Anticipated Operational Transient more severe.

COR0021402001060B Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor pressure

INT0060119001050A For each of the following Anticipated Operational Transient categories listed, select an example of each transient, as stated in the CNS USAR: Nuclear system pressure increase

COR0021402001060D Given a specific Main Steam system malfunction, determine the effect on any of the following: Reactor power

Skills

295020.AK3.04 Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.5 / 45.6) Reactor pressure response. (4.1/4.1)

Question 68   13752   (1 point(s))

During a LOCA, the CRS requests a round of primary containment parameters. The RO reports the following:

- DW pressure: 4.5 psig and rising
- DW temperature: 160°F and rising
- Torus water level: 0 inches
- Torus water temp: 85°F

Does the report made by the RO meet the requirement(s) for the requested information and why or why not?

a. Meets. **ONLY** values are required.

b. Meets. **ONLY** changing parameters require trends or rate of trend.

c. **NOT** meet. Trend and rate of trend is required for the identified parameters.

d. **NOT** meet. Trends are required for ALL of the identified parameters. Rate of trend is not required.

Answer 68

c. **NOT** meet. Trend and rate of trend is required for the identified parameters.

During execution of EOPs, the CRS is expected to periodically direct one CRO to provide a round of primary containment parameters. Upon request, the parameter, trend, and rate of trend are to be included with no exceptions. Per 2.0.3, step 7.3.7, only the parameter and trend are normally required but since a "round of parameters" was requested, the rate of trend is also required.

REFERENCE: 2.0.3, step 8.6

Foils:

a. Does not meet standards. Trends and rate of trends are required.
b. Does not meet standards. Trends and rate of trends are required for parameters that are stable.

d. Rate of trend is required.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

OTH0151003001010D From memory define the following terms in accordance with in procedure 2.0.3, Conduct of Operations, and Operations Instruction #7: Briefs and updates

Skills

2.4.12 Knowledge of general operating crew responsibilities during emergency operations. (CFR: 41.10 / 45.12)

Question 69   17873   (1 point(s))

Which one of the following is the FEDERAL LIMIT for annual whole body dose to a control room operator in a calendar year?

a. 4 rem/calendar year

b. 5 rem/calendar year

c. 15 rem/calendar year

d. 50 rem/calendar year

Answer 69

b. 5 rem/calendar year

The federal limit is 5 rem/year.

Reference: 10CFR20, 9.ALARA.1

Foils:

a. 4 Rem is a CNS administrative limit.
c. 15 Rem is the federal limit for dose to the lens of the eye.
d. 50 Rem is the federal limit for dose to the extremities.

New
Difficulty 2
Cognitive level: 1

Enabling Objectives

INT03201160000100 State the CNS administrative and federal exposure limits

Skills

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10)

Question 70   6047   (1 point(s))

A Site Area Emergency has been declared at CNS. Which Emergency Response Facility provides the location for the overall response to an emergency, once all the Emergency Response Facilities are activated?

a. Control Room

b. Technical Support Center.

c. Operations Support Center.

d. Emergency Operations Facility.

Answer 70

d. Emergency Operations Facility

Reference: 5.7.9

Foils:

a. The Control room manages the plant the EOF provides the overall response.
b. The TSS provides technical support to the EOF
c. The OSC provides support to the EOF.

Bank
Difficulty 2
Cognitive Level 1

Enabling Objectives

GEN0030401B0B040B Emergency Response Facilities (ERFs): b) Describe the functions of each of the following ERFs: 1) Control Room (CR) 2) Technical Support Center (TSC) 3) Operation Support Center (OSC) 4) Emergency Operations Facility (EOF) 5) Joint Information Center (JIC)

Skills

2.4.29 Knowledge of the emergency plan. (CFR: 43.5 / 45.11)

Question 71   3144   (1 point(s))

Given the following conditions:

- The plant is operating at 100% power.
- High Pressure Coolant Injection (HPCI) is in service for the quarterly full-flow test.
- Initial Suppression Pool Temperature is 92°F.

As HPCI continues to operate, what Technical Specifications actions are required?

When Suppression Pool Temperature *exceeds* …

a. 100°F, reduce thermal power to ≤ 1% Rated Thermal Power

b. 100°F, verify Suppression Pool temperature is less than 120°F once every 30 minutes.

c. 105°F, remove HPCI from service.

d. 105°F, place the Reactor Mode Switch in SHUTDOWN.

Answer 71

c. 105°F, remove HPCI from service.

REFERENCE: T.S 3.6.2.1

Foils:

a. Not required until completion times not satisfied.
b. Not required until 110°F
d. Not required until 110°F

Bank
Difficulty 2
Cognitive Level 2

Enabling Objectives

INT00705070010400 From memory, in MODES 1,2, and 3, state the actions required in less than one hour if suppression pool average temperature exceeds 105 degrees F and THERMAL POWER IS > 1% RAP and performing testing that adds heat to the suppression pool (LCO 3.6.2.1)

Skills

295013.AA2.01 Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.8/4.0)

Question 72   17855   (1 point(s))

With the plant operating at full power, what is the basis for the Technical Specification requirement to place the Reactor Mode Switch in SHUTDOWN on high suppression pool temperature?

a. Ensure that low pressure ECCS pump NPSH requirements are met during a design bases LOCA.

b. Ensure that low pressure ECCS pump Vortex limits re not exceeded during a design bases LOCA.

c. Ensures that unstable condensation (chugging) does not occur in the suppression pool during a design bases LOCA.

d. Ensures the reactor is shutdown to prevent the suppression pool from being heated beyond design limits by the steam generated during a design bases LOCA.

Answer 72

d. Ensures the reactor is shutdown to prevent the suppression pool from being heated beyond design limits by the steam generated during a design bases LOCA.

REFERENCE: T.S. 3.6.2.1 Bases

Foils:

a. NPSH is an EOP related concept, not directly a Tech Spec limit bases.
b. Vortex limits is an EOP related concept, not a Tech Spec limit bases.
c. Chugging is an EOP related concept, not a Tech Spec limit bases.

New
Difficulty 3
Cognitive Level 1

Enabling Objectives

INT00705070010500 From memory in MODES 1,2, and 3, state the actions required in less than one hour if suppression pool average temperature > 110 degrees F but is either less than or equal to 120 degrees F (LCO 3.6.2.1)

INT00705070010200 Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Skills

295026.EK3.05 Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.5 / 45.6) Reactor SCRAM. (3.9/4.1)

Question 73   2871   (1 point(s))

A loss of high pressure injection has resulted in both Core Spray subsystems automatically starting. NO leaks exist inside the drywell and all off-site power sources remain available. The following conditions exist:

- Drywell pressure	0.5 psig (stable)
- Reactor pressure	200 psig (stable)
- PMIS	Unavailable

What effect does Core Spray pump operation have on Suppression Pool level under these conditions?

a. Suppression Pool Wide Range Level Recorders, PC-LR-1A & 1B will indiate LOWER than ***actual*** level. The value must be corrected to implement EOP actions.

b. Suppression Pool Narrow Range Level Indicators, PC-LI-12 & 13 will indiate LOWER than ***actual*** level. The value does **NOT** need to be corrected to implement EOP actions.

c. Suppression Pool Wide Range Level Recorders, PC-LR-1A & 1B will indiate HIGHER than ***actual*** level. The value does **NOT** need to be corrected to implement EOP actions.

d. Suppression Pool Narrow Range Level Indicators, PC-LI-12 & 13 will indiate HIGHER than ***actual*** level. The value must be corrected to implement EOP actions.

Answer 73

a. Suppression Pool Wide Range Level Recorders, PC-LR-1A & 1B will indiate LOWER than ***actual*** level. The value must be corrected to implement EOP actions.

JUSTIFICATION: The SRO implements the EOPs and must know the actual value of Primary Containment water level to accurately assess plant conditions. The RO will operate Core Spray as necessary to maintain RPV water level, but takes no action without direction based on PC water level.

REFERENCE: Low Pressure Core Spray Text, 2.2.9; 2.6

Foils:

b. Narrow range is un-affected.
c. Wide Range Level Recorders will read low

d.	Narrow range is un-affected.

10CFR55.43(5)
Modified
Cognitive Level: 2
Difficulty: 3

Enabling Objectives

COR0020602001080E	Given a Core Spray component manipulation, predict and explain the changes in the following: Suppression Pool water level

COR0020602001030B	Describe the interrelationships between the Core Spray and the following: Suppression Pool

Skills

295030.EA2.01 Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Suppression pool level (4.1*/4.2*)

Question 74 17902 (1 point(s))

During a plant startup at 35% power, the following conditions exist:

- "B" Reactor Feedpump is in service
- "B" Narrow Range level transmitter failed upscale
- "A" Narrow Range level instrument is selected for RPV level control
- Reactor Vessel Level Control is in single element control

During investigation of the "B" level instrument by technicians, power is lost to only Reactor Vessel level Control components and associated high level trip circuitry powered from panel AA2.

(1) What is the maximum time operation is allowed by Technical Specifications at this power level with these failures without performing any additional actions and,
(2) What is the effect (if any) of restoring power to DC panel AA2 PRIOR TO performing any additional actions?

a. (1) 6 hours,
   (2) The main turbine and both Reactor Feedwater pumps **WILL** trip.

b. (1) 13 hours,
   (2) The main turbine and both Reactor Feedwater pumps **WILL** trip.

c. (1) 7 days, 4 hours,
   (2) The main turbine and both Reactor Feedwater pumps will **NOT** trip.

d. (1) There is no time limit per Technical Specifications,
   (2) The main turbine and both Reactor Feedwater pumps will **NOT** trip.

Answer 74

d. (1) There is no time limit per Technical Specifications,
   (2) The main turbine and both Reactor Feedwater pumps will **NOT** trip.

EXPLANATION: The failed upscale transmitter results in a loss of one channel of required high level trip circuitry (T.S. 3.3.2.2) and must be placed in trip within 7 days. Failed upscale results in a trip signal, so the action is already accomplished. The loss of AA2 also drops out the relay (resulting in a trip signal) that feeds into the high level trip circuitry for the RFPs and main turbine for the "B" channel. Upon restoration of power, there will still only be one channel tripped and the logic requires at least 2 channels tripped.

REFERENCE: GE 791E257, Feedwater Control System, GE 719E596BD, Feedwater Control System, COR0023202 student text

Foils:

a. This would be the time limit with a loss of trip capability. The RFP and main turbine will not trip.
b. This is the time limit for T.S. 3.0.3 which does not apply. The RFP and main turbine will not trip.
c. This is the time limit if the channel is not placed in the tripped condition.

10CFR55.43(2)
New
Cognitive Level: 3
Difficulty: 4

Provide to Candidate: Tech Spec LCO 3.3.2.2 and associated Bases

Enabling Objectives

COR0020702001080B Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)

COR0020702001080O Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor feedwater system

COR0023202001040A State the electrical power supplies to the following: RVLC Components

INT00705040010100 Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.

INT00705040010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

COR00207020010200 Given conditions and/or parameters associated with the DC Electrical Distribution System, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operation are met.

Skills

295004.AK1.06 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.8 to 41.10) Prevention of inadvertent system (s) actuation upon restoration of D.C. power. (3.3/3.6)

Question 75 17877 (1 point(s))

The plant was at power when the B Recirc Pump Controller output was slowly rising. The RO locked the scoop tube because the controller did NOT respond in manual. Reactor power is currently 100%. Loop A and Loop B Jet Pump Flow instruments indicate:

- NBI-FI-92A = 32 x 10^6 lbs/hr
- NBI-FI-92B = 37 x 106 lbs/hr

What is the TS 3.4.1 implication for this condition?

a. LCO Statement is MET because the 5% mismatch requirement that is applicable for these plant conditions is SATISFIED.

b. LCO Statement is NOT MET because the 5% mismatch requirement that is applicable for these plant conditions is EXCEEDED.

c. LCO Statement is MET because the 10% mismatch requirement that is applicable for these plant conditions is SATISFIED.

d. LCO Statement is NOT MET because the 10% mismatch requirement that is applicable for these plant conditions is EXCEEDED.

Answer 75

b. LCO Statement is NOT MET because the 5% mismatch requirement that is applicable for these plant conditions is EXCEEDED.

$\leq$5% of rated core flow is the surveillance criteria when core flow is $\geq$ 70% rated core flow. Jet Pump Flow mismatch must be $\leq$3.67 E6 lbs/hr (5%) at $\geq$51.45 E6 lbs/hr (70%). The current mismatch is outside this limit.

Reference: TS 3.4.1 and bases

Foils:

a. Jet Pump Flow mismatch of $\leq$5% is exceeded.

c. 10% mismatch does not apply, core flow is > 70%.

d. 10% mismatch does not apply, core flow is > 70%.

KA 295001 2.1.7 (4.4)

10CFR55.43(2)
10CFR55.43(5)
New
Cognitive level: 3
Difficulty: 3

Provide to the candidate: T.S. 3.4.1, bases & 6.LOG.601 Attachment 12

Enabling Objectives

INT00705050010100 Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.

INT00705050010200 Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.4 Specification.

Skills

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)

Question 76   2255   (1 point(s))

The plant is operating at rated power when the Normal Transformer secondary voltage instantaneously drops to 3700 VAC. 345 KV, 161 KV and 69 KV voltages remain normal.

How will the plant respond to this voltage reduction over the next 1 (one) minute?
(ASSUME NO Operator Action)

Buses 1F and 1G will transfer to the . . .

a.   Diesel Generators. The reactor **WILL** scram.

b.   Diesel Generators. The reactor will **NOT** scram.

c.   Emergency Transformer. The reactor **WILL** scram.

d.   Emergency Transformer. The reactor will **NOT** scram.

Answer 76

d.   Emergency Transformer. The reactor will **NOT** scram.

With bus 1A and 1B powering 1F and 1G and both receiving power from the normal transformer, the low NSST secondary voltage will be felt on the emergency busses. 12.5 seconds after voltage lowers below 3880 VAC, breakers 1FA and 1GB will trip. This will cause a loss of voltage to be sensed on the emergency busses. This will start the EDGs. Breakers 1FS and 1GS will close to supply the emergency busses from the emergency transformer as it is at rated voltage. The transfer will occur in ~ 1 second. The RPS MG sets have a 1.2 second time delay that will reclose the motor starter if power is restored in < 1.2 seconds. RPS voltage and hertz will be maintained by the flywheel during the 1 second, so RPS will not be lost. No normal busses will trip and with no RPS trip, the reactor will not scram.

REFERENCE : 2.2.18 Attachment 22, 2.2.22 Attachment 1, 5.3GRID discussion. SOER 99-01

Foils:

a.   The Diesels will start, but the Emergency Transformer is available and will supply the busses. The RPS MG sets remain energized.

b. The Diesels will start, but the Emergency Transformer is available and will supply the busses.
c. The RPS MG sets remain energized.

10CFR55.43(5)
Bank
Cognitive Level: 3
Difficulty: 3

Enabling Objectives

COR0010102001080B Predict the consequences of the following on plant operation: 4160V Critical bus undervoltage

COR0010102001090G Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Transfer from preferred power to alternate power supplies

COR0010102001090C Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Automatic bus transfer

COR0010102001090B Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Circuit breaker automatic trips

COR0010102001130K Predict the consequences of the following events on the AC Electrical Distribution System: Degraded system voltages

Skills

295003.AK1.03 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10) Under voltage/degraded voltage effects on electrical loads. (2.9/3.2)

Question 77 16421 (1 point(s))

With the plant at 100% power, the following occurred on the "B" Reactor Protection System (RPS):

- The upstream RPS Electrical Protection Assembly (EPA) EPA logic card failed, causing the breaker to trip.
- The downstream RPS EPA remained closed.
- The plant response to the "B" RPS power loss was per design.
- The "B" RPS distribution panel was switched to the alternate power supply.
- The half scram and isolations were reset.

(1) What is the operational status of the "B" RPS MG Set EPA breakers and,
(2) Is "B" RPS bus operable by TS after it is transferred to the alternate supply?

a. (1) **Both** "B" RPS MG EPA breakers are inoperable,
(2) YES

b. (1) **Both** "B" RPS MG EPA breakers are inoperable,
(2) NO

c. (1) **Only** one of the "B" RPS MG EPA breakers are inoperable,
(2) YES

d. (1) **Only** one of the "B" RPS MG EPA breakers are inoperable,
(2) NO

Answer 77

a. (1) **Both** "B" RPS MG EPA breakers are inoperable,
(2) YES

REFERENCE:
TS 3.3.8.2, LCO Statement
TS 3.3.8.2 Bases

Distracter b: "B" RPS is operable.
Distracter c: Both RPS EPAs are inoperable.
Distracter d: Both RPS EPAs are inoperable and "B" RPS is operable

10CFR55.43(2)
Bank
Cognitive Level: 2
Difficulty: 3

Enabling Objectives

COR00221020010200 Given a condition(s) and/or parameters associated with the RPS, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operations are met.

COR0022102001040A Describe the RPS design features and/or interlocks that provide for the following: System redundancy and reliability

COR0022102001050C Briefly describe the following concepts as they apply to RPS: EPA operation

Skills

212000.A4.07 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System status lights and alarms (4.0*/3.9*)

Question 78   17887   (1 point(s))

What is the bases for the the refueling interlocks associated with the reactor mode switch REFUEL position?

a. To prevent criticality during refueling by ensuring that fuel assemblies are not loaded into the core with any control rod withdrawn.

b. To prevent control rod/fuel assembly configurations during refueling from resulting in fuel enthalpy above 280 cal/gm by preventing multiple control rod withdrawal.

c. To ensure that fuel assembly loading sequence and configurations are restricted to maintain an adequate shutdown margin by monitoring control rod position and hoist load status.

d. To ensure that radioactivity releases as a result of a refueling accident are maintained below a small fraction of 10 CFR 100 limits by preventing more that one (1) hoist being loaded at the same time.

Answer 78

a. To prevent criticality during refueling by ensuring that fuel assemblies are not loaded into the core with any control rod withdrawn.

REFERENCE: Tech Spec 3.9.1 Bases

Foils:

b. 280 cal/gm is a RWM bases (N/A during refueling). The refueling interlocks place no restrictions on fuel assembly configuration.
c. The refueling interlocks do not ensure an adequate shutdown margin.
d. The refueling interlocks do not prevent loading of multiple hoists at the same time and the loading of the hoists is not related to the 10CFR100 release rates.

10CFR55.43(2)
10CFR55.43(7)
New Question
Cognitive level: 1
Difficulty: 3

Enabling Objectives

INT00705100010200 Discuss the applicable Safety Analysis in the Bases associated with each Section 3.9 Specification.

Skills

295023.AK1.03 Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: (CFR: 41.8 to 41.10) Inadvertent criticality. (3.7/4.0)

Question 79 16441 (1 point(s))

Refueling operations are in progress with the reactor vessel head removed and a partial load of fuel is in the vessel. Shutdown margin check has been performed.

Which one of the following is a CORE ALTERATION?

a. Installing a control rod blade into an empty cell.

b. Driving a Source Range Monitor detector to full in.

c. Performing a friction test on a control rod in a loaded cell.

d. Inserting the LPRM Instrument Handling Tool below the top guide.

Answer 79

c. Performing a friction test on a control rod in a loaded cell.

REFERENCE: TS 1.1: CORE ALTERATION definition

Justification: Core alteration includes movement of any reactivity controlling component with the exceptions specified: SRM movement is an exception, control rod movement if there is no fuel in the associated core cell is an exception. The LPRM instrument handling tool is not a reactivity control component.

Distracter a: Control rod movement provided there are no fuel assemblies in the associated core cell is not considered to be a CORE ALTERATION.
Distracter b: Movement of a SRM is not considered to be a CORE ALTERATION.
Distracter d: The LPRM instrument handling tool is not a reactivity control component.

10CFR55.43(2)
10CFR55.43(7)
Bank
Cognitive Level: 1
Difficulty: 2

Enabling Objectives

INT00705010010300 Define the terms listed in CNS Technical Specification 1.1.

Skills

2.2.26 Knowledge of refueling administrative requirements. (CFR: 43.5 / 45.13)

Question 80 119 (1 point(s))

The plant is operating near rated power when Recirculation pump A speed control malfunctions, requiring the scoop tube control to be locked out. Jet pump loop flows are mismatched by 12%.

What is the safety significance of operating the Reactor Recirc Pumps with this amount of flow mismatch?

Reactor Recirculation Pump operation with this much flow mismatch will result in ...

a. jet pump failure.

b. regional power oscillations.

c. non-conservative APRMs flow referenced scram setpoints.

d. operation outside conditions assumed in the LOCA analysis.

Answer 80

d. operation outside conditions assumed in the LOCA analysis.

REFERENCE: T.S.3.4.1

Foils:

a. Not a safety basis for mismatch limits.
b. Not a safety basis for mismatch limits.
c. Not a safety basis for mismatch limits.

10CFR55.43(2)
Bank
Cognitive Level: 1
Difficulty: 3

Enabling Objectives

INT00705050010200 Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.4 Specification.

Skills

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2)

202002.A2.04 Ability to (a) predict the impacts of the following on the RECIRC FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct...: (CFR: 41.5 / 45.6) Recirculation pump speed mismatch between loops: Plant Specific (3.0/3.2)

Question 81   16632   (1 point(s))

The plant is at 55%, raising power following a reactor startup. At 08:00, B-1/E-1 BPV FAST-OPEN PERMISSIVE INHIBIT, alarms. The cause of the alarm CANNOT be corrected for 24 hours.

What is the **MINIMUM** action (if any) required to restore compliance with Technical Specifications within the 24 hour period?
(NOTE: The choices are listed from MINIMUM to MAXIMUM action order.)

a. No actions are required.

b. Transfer Bypass valves to MANUAL.

c. Adjust MCPR limits for inoperable bypass valves.

d. Reduce power to below 25% RTP.

Answer 81

d. Reduce power to below 25% RTP.

Per TS 3.7.7 LCO statement, The Main Turbine Bypass System shall be OPERABLE O<u>R</u> LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for one inoperable main turbine bypass valve, as specified in the COLR, are made applicable. Since all 3 bypass valves are inoperable, power must be reduced to exit applicability.

REFERENCES: TS 3.7.7; LCO Statement, Action A.1 and B.1

Distractors:
a - no requirement for this
b - no requirement for this
c - will not address multiple bypass valves inoperable

K/A 241000 (Pressure Regulating System) 2.2.22 (4.1)

10CFR55.43(2)

Bank
Cognitive Level: 2

Difficulty: 2

Provide to Candidate: TS 3.7.7

Enabling Objectives

INT00705080010100 Given a set of plant conditions, recognize non-compliance with a Chapter 3.7 LCO.

INT00705080010300 Given a set of plant conditions that constitutes non-compliance with a Chapter 3.7 LCO, determine the ACTIONS that are required.

Skills

2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2)

Question 82 17895 (1 point(s))

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

- A degraded grid condition exists
- All off-site source voltage values are currently at the minimum for operability
- The main generator is at + 350 MVAR
- The main generator is currently 150 MVAR above the limit set by the load dispatcher.

What effect (if any) will a main generator trip have on off-site source operability?

a. **BOTH** off-site sources remain OPERABLE.

b. **BOTH** off-site sources become **IN**OPERABLE.

c. The Emergency Transformer will be **IN**OPERABLE; the Startup Transformer will be OPERABLE.

d. The Emergency Transformer will be OPERABLE; the Startup Transformer will be **IN**OPERABLE.

Answer 82

b. **BOTH** off-site sources become **IN**OPERABLE.

The 345 KV system is connected to the 161 KV system at CNS and is connected to the 69 KV at Brock. Plant experience has shown that CNS VAR changes directly and immediately affect 69 and 161 KV voltages. 5.3GRID step 5.12 states "Operating the generator MVAR output in accordance with Doniphan's direction is required to ensure that the analysis used to establish operability of the off-site power sources remains valid."

REFERENCE: 5.3GRID, TS 3.8.1 Bases, Electrical Theory, NRC IN 2000-06

Foils:

a, c, d **BOTH** off-site sources become **IN**OPERABLE.

10CFR55.43(2)
10CFR55.43(5)

New
Cognitive Level: 3
Difficulty: 3

Enabling Objectives

COR0010102001130A   Predict the consequences of the following events on the AC Electrical Distribution System: Turbine/generator trip

INT00705090010100   Given a set of plant condtions, recognize non-compliance with a Section 3.8 LCO.

Skills

262001.K6.03   Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: (CFR: 41.7 / 45.7) Generator trip (3.5/3.7)

Question 83   121   (1 point(s))

The RCIC turbine has been disassembled for overhaul during the refueling outage. Following the outage, plant management has requested unit startup be commenced. Maintenance informs the control room that the RCIC turbine re-assembly is almost complete and the post maintenance testing will take several hours.

Which of the following is required by Technical Specifications?

a. The plant **CAN** enter MODE 3 but **CANNOT** enter MODE 2 **AND CANNOT** exceed 150 psig.

b. The plant **CANNOT** enter MODE 3 **OR** MODE 2 until RCIC is re-assembled and all testing that can be performed ***prior*** to startup is completed.

c. The plant **CAN** enter MODE 2 provided the RCIC System re-assembly and post maintenance testing is complete ***prior*** to exceeding 150 psig.

d. The plant **CAN** enter MODE 2 provided the RCIC System re-assembly and post maintenance testing is complete ***within 24 hours*** of exceeding 150 psig.

Answer 83

c. The plant **CAN** enter MODE 2 provided the RCIC System re-assembly and post maintenance testing is complete ***prior*** to exceeding 150 psig.

REFERENCE: TS SR 3.0.4, TS 3.5.3

Foils:

a. The plant can enter Mode 2. The LCO is not applicable until 150 psig.
b. The plant can enter Mode 2 and Mode 3. The LCO is not applicable until 150 psig.
d. The 24 hours is for completing surveillances for a known operable RCIC, not an exception to applicability.

10CFR55.43(2)

10CFR55.43(3)
Bank
Cognitive Level: 3
Difficulty: 3

Provide to Candidate: T. S. 3.5.3 and associated bases

Enabling Objectives

INT00705060010100 Given a set of plant conditions, recognize non-compliance with a Section 3.5 LCO.

INT00705060010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.5 LCO, determine the ACTIONS that are required.

INT00705010010200 Given a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Skills

2.1.12 Ability to apply technical specifications for a system. (CFR: 43.2 / 43.5 / 45.3)

Question 84   16446   (1 point(s))

During the channel calibration for one reactor pressure high RPS instrument, the as-found and as-left values for one instrument are 1055 psig (as-found *with* head correction) and 1039 psig (as-left *with* head correction).

What is the operability of this instrument <u>upon learning the as-found value</u> and <u>after completing the calibration</u>?

With the as-found value, it was __________.   After calibration, it is ____________.

a.	**IN**operable	operable
b.	**IN**operable	**IN**operable
c.	operable	operable
d.	operable	**IN**operable

Answer 84

a.	**IN**operable	operable

NOTE: The surveillance has a 13# head correction to the values collected on the data sheet for operability assessment.

REFERENCE:
TS 3.3.1.1 Table 3.3.1.1, function 3, Allowable Value
TS Bases 3.3.1.1, Background page 3, Paragraph 2

Foils:

b. Became operable when it was calibrated to below the allowable value.
c. When the allowable value is exceeded, the instrument is inoperable.
d. When the allowable value is exceeded, the instrument is inoperable. became operable when it was calibrated to below the allowable value.

10CFR55.43(2)

Bank
Cognitive Level: 1
Difficulty: 2

Enabling Objectives

COR0022102001050A Briefly describe the following concepts as they apply to RPS: Logic arrangements

INT00705040010100 Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.

COR0022102001120A Given plant conditions determine if: A Full Scram should have occurred

COR0022102001120B Given plant conditions determine if: A Half Scram should have occurred.

Skills

2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)

Question 85   17894   (1 point(s))

The plant is at 100% when high area radiation in the Reactor Building occurs. Current conditions are:

- RCIC supply pressure   900 psig
- HPCI supply pressure   990 psig
- Torus Area temperatures   185°F and rising
- Reactor Building ventilation exhaust radiation levels are 7 mr/hr and rising
- CRD hydraulic equipment area (North) radiation levels are 90 mr/hr and rising

What is the source of the high area radiation?

a. RCIC

b. HPCI

c. RWCU

d. North Scram Discharge Volume

Answer 85

a. RCIC

REFERENCE: EOP-5A

Foils:

b. RCIC pressure is less than Rx pressure, HPCI is not.
c. RWCU would not bring in the 903 alarms or cause HPCI/RCIC pressure difference.
d. SDV has no pressure on it with no scram and would not cause cause HPCI/RCIC pressure difference.

10CFR55.43(5)

New
Cognitive Level: 2
Difficulty: 2

Enabling Objectives

INT00806170010600 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Skills

295033.EA2.03 Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: (CFR: 41.10 / 43.5 / 45.13) ?Cause of high area radiation. (3.7/4.2)

Question 86   17903   (1 point(s))

A Loss of Coolant Accident has occurred with the following conditions:

-	Reactor pressure	700 psig
-	Drywell pressure	12 psig
-	Drywell temperature	335° F
-	Primary Containment water level	17.3 feet
-	Torus temperature	187° F

What is a potential consequence of initiating Drywell spray under these conditions?

IF Drywell Spray is initiated, . . .

a. the vent header downcomers may be damaged due to excessive flow.

b. the Torus-to-Drywell Vacuum Breakers may be damaged due to excessive flow.

c. the primary containment may be damaged due to excessive torus/drywell differential pressure.

d. the Reactor Building-to-Torus Vacuum Breakers may be damaged due to excessive torus/reactor building differential pressure.

Answer 86

c. the primary containment may be damaged due to excessive torus/drywell differential pressure.

REFERENCE: INT0080613, PSTG, EPG

Justification: Torus-to-drywell vacuum Breakers are submerged at 16.5' and will not pass sufficient flow to the Drywell, resulting in excessive differential pressure.

Foils:
a. excessive flow is not a problem.
b. excessive flow is not a problem.
d. The excessive differential pressure is from torus to drywell, not torus to Rx bldg.

10CFR55.43(5)
New
Cognitive Level: 2

Difficulty: 2

Provide to Candidate: EOP 3A with entry conditions removed.

Enabling Objectives

INT0080613001040C State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A

INT00806130011200 Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Skills

295029.EK2.05 Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: (CFR: 41.7 / 45.8) Containment/drywell vacuum breakers. (3.1/3.3)

Question 87   16519   (1 point(s))

A LOCA has occurred and the following conditions exist:

- Drywell H_2 9% (rising slowly)
- Drywell O_2 4% (steady)
- Suppression Chamber H_2 6% (rising slowly)
- Suppression Chamber O_2 3%. (steady)

What is the status of the Primary Containment H_2/ O_2 combustible limit (above or below the limit) and required actions (emergency depressurization or primary containment venting)?

a. BELOW the limit. Emergency depressurization is required.

b. BELOW the limit. Emergency venting of the Primary Containment is required.

c. ABOVE the limit. Emergency depressurization is required.

d. ABOVE the limit. Emergency venting of the Primary Containment is required.

Answer 87

b. BELOW the limit. Emergency venting of the Primary Containment is required.

Justification: The limits, 6%, H_2 and 5%, O_2 in either the torus or drywell are the limits for the primary containment. These indications are high for H2 but lack sufficient O2 for Combustiblity. SAP 5.9H2O2 requires emergency venting the PC.

REFERENCE: EOP-3A; PC/G-2 and TABLE 7.
SAP5.9H202

Foils:

a. ED is not required.
c. Below limit, ED is not required.
d. Below limit.

10CFR55.43(5)

Bank
Cognitive Level: 2
Difficulty: 3

Provide to Candidate: EOP-3A with Cautions and entry conditions removed

Enabling Objectives

INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

INT00806070010100 List conditions that require RPV Emergency Depressurization.

Skills

500000.EA2.03 Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.10 / 43.5 / 45.13) Combustible limits for drywell (3.3/3.8)

Question 88   17904   (1 point(s))

The plant was operating at full power when the MSIVs closed due to a main steam line break. Many control rods failed to insert. The following conditions exist:

-	Reactor water level	TAF to +50" (corrected fuel zone)
-	Reactor pressure	800 to 1000 psig (SRVs)
-	Reactor power	oscillating 22 to 38%
-	Primary Containment water level	13.5 feet
-	Torus water temperature	106 °F
-	No attempt has been made to initiate Boron injection	

What is a potential consequence (if any) of delaying boron injection until now and why/why not?

a. There is no potential consequence as no limit or graph has been exceeded.

b. The power peaks may result in severe core damage due to exceeding the Large Oscillation Threshold.

c. The reactor may not be shutdown before the Heat Capacity Temperature Limit (Graph 7) is exceeded due to the assumptions used for the Boron Injection Initiation Temperature (Graph 8).

d. A large break LOCA may result in the Primary Containment Pressure Limit (Graph 11) being exceeded due to torus temperature being higher than the value assumed in the accident analysis.

Answer 88

c. The reactor may not be shutdown before the Heat Capacity Temperature Limit (Graph 7) is exceeded due to the assumptions used for the Boron Injection Initiation Temperature (Graph 8).

EXPLANATION: The "flat" part of BIIT (110°F) is based on the Tech Spec required shutdown. The "sloped" part adjoining the "flat" part is based on the injection rate of the SLC pumps and getting HSBW injected prior to HCTL. Injecting boron just below the "flat" part of the curve does NOT ensure HSBW is injected prior to HCTL.

REFERENCE: INT0080606, PSTG

FOILS:

a. BIIT is not exceeded, but there is a potential consequence based on the curve assumptions.
b. LOT is 25% peak to peak. Current oscillations are 16%. The LOT is one of the basis for injecting SLC, but it is not met.
d. This is one of the basis for HCTL.

10CFR55.43(2)
10CFR55.43(5)
New
Cognitive Level: 3
Difficultly: 4

Enabling Objectives

INT00806060010800 Explain the basis for injecting boron before the Boron Injection Initiation Temperature is exceeded and when large periodic neutron flux oscillations in excess of 25% occur.

Skills

295037.EK3.02 Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.5 / 45.6) SBLC injection (4.3*/4.5*)

Question 89   8930   (1 point(s))

The plant is operating at 100% power. HPCI is tagged out due to auxiliary oil pump motor replacement. A loss of off-site power occurs.

The following conditions exist:

- RPV water level   - 115" (Wide Range, lowering).
- RCIC   Isolated on high steam flow.
- Reactor pressure 920 psig (rising slowly).
- Torus Temperature   165 °F
- Torus Level   +2" (narrow range, rising).
- Drywell Temperature   270 °F (lowering slowly).
- Control Rods   All but one fully inserted.

What actions are required per EOP Flowchart 1A, RPV CONTROL?

a. Exit EOP Flowchart 1A and enter Flowchart 2B.

b. Restore RPV water level +3 to +54" following automatic ADS actuation.

c. Remain in the pressure leg of EOP Flowchart 1A. Inhibit ADS and attempt to restore RPV water level +3 to +54" with CRD and SLC.

d. Exit the pressure leg of EOP Flowchart 1A and enter EOP Flowchart 2A. Inhibit ADS and attempt to restore RPV water level +3 to +54" with CRD and SLC.

Answer 89

c. Remain in the pressure leg of EOP Flowchart 1A. Inhibit ADS and attempt to restore RPV water level +3 to +54" with CRD and SLC.

Foils:

Note: DGs are available there is only a loss of off-site power.

a. RPV level in known. Entering Flowchart 2B is not required.
b. ADS is inhibited in Flowchart 1A, step RC/L-5.
c. Entering Flowchart 2A is not required because ED is not require.

10CFR55.43(5)
Bank
Cognitive Level: 2
Difficultly: 3

Provide to Candidate: EOP-1A

Enabling Objectives

INT00806090011100 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Skills

295031.EA1.13 Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) Reactor water level control. (4.3*/4.3*)

Question 90   16569   (1 point(s))

A LOCA has occurred with the following conditions:

- PMIS is unavailable
- All control rods are fully inserted
- RHR pumps A, B, and D are NOT available
- HPCI and RCIC are NOT available
- RHR pump C in the LPCI mode is being used to maintain RPV level
- RPV water level is +20" (corrected fuel zone) (rising 10 inches per minute)
- There is a leak on the "B" RHR pump suction from the torus, next to the torus.

- RHR flow rate is 8500 gpm (maximum available)
- Torus pressure is 4 psig (stable)
- Torus average water temp is 185°F (rising slowly)
- Primary Containment water level is 7 feet (stable)

The SS directs that NPSH and Vortex requirements be complied with.

What are the **MINIMUM** actions (if any) necessary to comply with the NPSH limits?
(NOTE: Choices are arranged in MINIMUM to MAXIMUM order.)

a. Maintain RHR flow at the current rate.

b. Reduce RHR flow to 6500 gpm.

c. Reduce RHR flow to 5000 gpm.

d. The RHR pump must be secured.

Answer 90

b. Reduce RHR flow to 6500 gpm.

With 4 psig overpressure and 3 feet of water above the suctions there is 5.29 psig overpressure requiring that flow be reduced to no more than 7000 gpm for NPSH concerns. Flow is also in the unsafe region of the vortex limit curve, but this curve is less limiting than the NPSH curve for this case. 10 inches per minute RPV level rise means that an excess flow of 1500 to 2000 gpm is available and flow could be reduced to 6500 gpm and still maintain RPV level above TAF.

JUSTIFICATION AS SRO KNOWLEDGE: With PMIS unavailable and elevated suppression pool water temperatures, the NPSH curves must be assessed manually. The Reactor Operator does not have the curves to assess, so the NPSH assessment must be performed by the CRS.

REFERENCE: EOP Graph 5

Foils:

a. Correct if wrong (10#) curve is used.
c. Correct if wrong curve (CS) is used.
d. Correct if wrong (0#) curve is used.

10CFR55.43(5)
Bank
Cognitive Level: 3
Difficulty: 4

Provide to Candidate: Full-size EOP NPSH/Vortex Graphs (Graph 4 and Graph 5), EOP Graphs

Enabling Objectives

INT00806180010300 Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

INT00806180010200 For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Skills

295026.EK1.01 Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.8 to 41.10) Pump NPSH. (3.0/3.4)

Question 91   16469   (1 point(s))

What is the basis for restarting building ventilation in the Turbine Building when executing EOP-5A, RADIOACTIVITY RELEASE CONTROL?

Operation of Turbine Building ventilation . . .

a. maintains equipment availability **AND** assures that radioactivity plates out evenly on horizontal surfaces.

b. maintains equipment availability **AND** assures that radioactivity releases pass through a monitored release point.

c. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

d. preserves personnel accessibility **AND** assures that radioactivity plates out evenly on horizontal surfaces.

Answer 91

c. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

REFERENCE: PSTG/SATG (EOP Bases)

Foils:

a. Turbine Building ventilation does not do either.
b. Turbine Building ventilation does not maintain equipment availability.
d. Turbine Building ventilation does not ensure radioactivity plates out evenly.

KA 295017 (High Off-site Release Rate) 2.4.48 (3.8)

10CFR55.43(4)
New
Cognitive level: 1
Difficulty: 3

Enabling Objectives

INT00806170010700 Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Skills

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. (CFR: 43.5 / 45.12)

Question 92   6090   (1 point(s))

A general Emergency has been declared with the following conditions:

-	wind direction	from 000º
-	Wind speed	15 mph
-	Stability class	D
-	Outside air temperature	60ºF
-	projected TEDE dose	0.15 rem
-	projected CEDE thyroid dose	0.42 rem

What Protective Action Recommendations are required at this time?

Recommend evacuation for __________________________.   Recommend remaining areas go indoors and monitor EAS/EBS.

a. two (2) mile radius **AND** sectors H, K **AND** J out to the five (5) mile radius

b. two (2) mile radius **AND** sectors A, B **AND** R out to the five (5) mile radius

c. two (2) mile radius **AND** sectors A, B **AND** R out to the ten (10 mile radius

d. five (5) mile radius **AND** sectors H, K **AND** J out to the ten (10) mile radius

Answer 92

a. two (2) mile radius **AND** sectors H, K **AND** J out to the five (5) mile radius

REFERENCE: EPIP 5.7.20, page 2, section 8.1.1.1

Foils:

b. Opposite wind direction.
c. Opposite wind direction and beyond 5 mile.
d. Wrong distances.

10CFR55.43(4)
Bank
Cognitive Level: 2
Difficulty: 3

Provide to Candidate: Proc 5.7.20

Enabling Objectives

GEN0030401D0D0600 State the Protective Action Guides used to determine the need to evacuate the general public.

Skills

2.4.44 Knowledge of emergency plan protective action recommendations. (CFR: 43.5 / 45.11)

295038.EK2.05 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: (CFR: 41.7 / 45.8) ?Site emergency plan. (3.7/4.7*)

Question 93 17883 (1 point(s))

While executing EOP-1A, RPV Control, how many OVERRIDES are to be under consideration if currently executing Step RC/L-12?

a. 1

b. 2

c. 3

d. 4

Answer 93

c. 3

3 overrides are applicable.

Reference: Volume 5-EOP-PWG: Section 4.7.3
EOP-1A

KA 2.4.6 (4.0)

10CFR55.43(5)
New Question
Cognitive level: 1
Difficulty: 2

Provide to Candidate: EOP-1A

Enabling Objectives

INT00806090011000 Identify any EOP support procedure addressed in Flowchart 1A and apply any associated special operating instructions or cautions.

INT00806090011100 Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

INT00806090010900 State the action required when adequate core cooling cannot be assured by core submergence or steam cooling.

Skills

2.4.6 Knowledge symptom based EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Question 94 17878 (1 point(s))

With the plant at 100% power, a small fire is reported in the HPCI room.

Which one of the following will require the declaration of an ALERT?

a. HPCI gland seal condenser motor is destroyed.

b. HPCI room temperature is 185°F and rapidly rising.

c. The fire is unable to be extinguished within 5 minutes.

d. The fire brigade is unable to respond within 10 minutes.

Answer 94

b. HPCI room temperature is 185°F and rapidly rising.

Per 5.4.1, General Fire Procedure, Step 4.2.2, the SS is to determine if declaration of an EAL per Procedure 5.7.1 is appropriate. The fire has the potential to affect safety related equipment so an alert is declared. At 195°F, the MAX SAFE temperature for the HPCI room is exceeded affecting HPCI operability and HPCI will automatically isolate.

Reference: 5.7.1: Attachment 1, section 5.2.1 and bases
5.4.1: 4.2.2

Foils:

a. Gland seal exhauster does not affect HPCI operability.
c. If the fire is unable to be extinguished within 10 minutes, then a NOUE is required. Only 5 minutes have expired so EAL is not required yet.
d. If the fire brigade is unable to respond within 10 minutes then it could be assumed that the fire has not been extinguished within 10 minutes. However, only a NOUE is required, not an alert.

KA 295032 2.4.27

10CFR55.43(5)

New
Cognitive level: 2
Difficulty: 3

Provide to Candidate: 5.7.1: Attachment 1 (EAL Matrix)

Enabling Objectives

GEN00301020000300 Cite the "Key Phrase" which describes the ALERT classification

Skills

2.4.27 Knowledge of fire in the plant procedure. (CFR: 41.10 / 43.5 / 45.13)

Question 95   17882   (1 point(s))

Per 5.7.14, Stable Iodine Thyroid Blocking (KI), who has the authority to designate when and to whom Potassium Iodide (KI) shall be distributed?

a. Plant Manager.

b. Emergency Director.

c. Chemistry Coordinator.

d. Radiological Control Manager.

Answer 95

b. Emergency Director.

Only the Emergency Director, normally acting on the recommendations of the Radiological Control Manager or Chem/RP Coordinator, may designate when and to whom KI shall be distributed.

REFERENCE: 5.7.14: 4.1

Foils:

a. Plant Manager does not authorize KI distribution.
c. Chemistry Coordinator makes recommendations but cannot authorize KI distribution.
d. Radiological Control Manager makes recommendations but cannot authorize KI distribution.

10CFR55.43(4)
New
Cognitive Level: 1
Difficulty: 2

Enabling Objectives

GEN00301030000200 State who, by title, may recommend to the emergency director when and to whom he may distribute potassium iodide

Skills

2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10)

Question 96 5735 (1 point(s))

The plant has been shutdown due to a forced outage. Entry into the drywell is required for performance of maintenance during the outage.

Which of the following are required for the ***initial*** drywell entry?

1. Reactor coolant temperature < 212°F.
2. Authorization from the Plant Manager logged in SS log.
3. Airborne particulate levels < 1000 cpm.
4. Drywell entry Hazardous Work Permit is issued.
5. Authorization from Primary Containment Coordinator.
6. Drywell O_2 concentration > 19.5%.

a. 1, 2, 3

b. 2, 3, 4

c. 3, 4, 5

d. 4, 5, 6

Answer 96

d. 4, 5, 6

REFERENCE: PR 2.0.10, section 3 and 5.

FOILS:
1. Not required.
2. Not required when shutdown (is required at power).
3. must be < 5000 cpm, not < 1000 cpm.

Bank
Cognitive level: 1
Difficulty: 3

Enabling Objectives

SKL0110101001380A 2.0.10, Primary Containment Access Control: Discuss the following as described in Conduct of Operations Procedure 2.0.10, Primary Containment Access Control: 1) Precaution and Limitations 2) Initial Primary Containment entry during power operations 3) Initial Drywell entry after shutdown 4) Drywell closeout 5) Initial Torus entry after shutdown 6) Torus closeout.

Skills

2.1.2 Knowledge of operator responsibilities during all modes of plant operation. (CFR: 41.10 / 45.13)

Question 97   16476   (1 point(s))

An RO left shift work on 6/3. The RO worked all scheduled workdays this year as BOP until leaving shift. Since leaving shift, the RO performed the following shifts as the BOP:

- 7/19 worked 12 hours
- 8/18 worked 12 hours
- 8/30 worked 12 hours
- 9/10 worked 12 hours

What is this operator's license status on 9/11?

a. License became inactive on 6/30.

b. License became inactive on 7/31.

c. License is active and only stays active with another 12-hour shift before 10/1.

d. License is active and only stays active with another 12-hour shift before 11/1.

Answer 97

c. License is active and only stays active with another 12-hour shift before 10/1.

Justification: Maintenance of an active license requires that five 12-hour shifts be stood per calendar quarter. As long as one more 12-hour shift is stood before the end of September (before 10/1/01) the license remains active.

REFERENCE: 10CFR55

Foils:

a. License is still active.
b. License is still active.
d. Calendar quarter ends September 30, not October 31.

10CFR55.43(1)
10CFR55.43(2)
Bank

Cognitive Level: 2
Difficulty: 2

Enabling Objectives

SKL00801020000600 State who is authorized to manipulate reactor and balance of pant controls.

Skills

2.1.10 Knowledge of conditions and limitations in the facility license. (CFR: 43.1 / 45.13)

Question 98 16462 (1 point(s))

Which one of the following documents contains the direction for the Shift Supervisor to indicate that a Temporary Configuration Change (TCC) has been logged in the TCC Log prior to installation?

a. Test Order

b. Caution Order

c. Clearance Order

d. Maintenance Work Order.

Answer 98

d. Maintenance Work Order.

The MWO shall include a sign-off for the Shift Supervisor (SS) indicating that the TCC has been logged into the TCC Log prior to installation.

REFERENCE: 3.4: 4.1.4.3

Foils:

a. Test Order does not contain the sign-off.
b. Caution Order does not contain the sign-off.
c. Clearance Order does not contain the sign-off.

10CFR55.43(3)
New Question
Cognitive Level: 1
Difficulty: 2

Enabling Objectives

INT032010300A010F Discuss the following as described in Conduct of Operations Procedure 2.0.1, Plant Operations Policy: System Review and Modification

Skills

2.2.11 Knowledge of the process for controlling temporary changes. (CFR: 41.10 / 43.3 / 45.13)

Question 99 17881 (1 point(s))

In preparation for refueling, maintenance informs you that the monorail hoist load cell on the refueling bridge must be replaced. The estimated time to replace and calibrate the load cell is 12 hours. Refueling is scheduled to start as soon as 6.REFUEL.304, Refueling Interlocks Functional Test is complete (in about 4 hours).

Which one of the following is the appropriate response to the monorail hoist load cell problem while trying to maintain schedule compliance?

a. Defer (NA) the monorail hoist checks because the monorail hoist is not needed for refueling.

b. Defer (NA) the monorail hoist checks because the load cell does not need to be operable for moving the fuel assemlies with the monorail hoist.

c. The monorail hoist load cell must be replaced and tested since it is required to move the fuel assemlies.

d. The monorail hoist load cell must be replaced and tested since it is required to move the blade guides.

.

Answer 99

a. Defer (NA) the monorail hoist checks because the monorail hoist is not needed for refueling.

Reference: 6.REFUEL.304, step 4.6 note

Refueling interlock checks are only necessary for the equipment used during the actual fuel movements, which include the refueling bridge/trolley and main hoist and the associated support systems such as air. The monorail hoist is not used to move nuclear fuel and does not have to be tested.

Foils:

b. The frame-mounted hoist is not used to move nuclear fuel and does not have to be tested.

c. The monorail hoist can be used to move nuclear fuel, but is not currently identified in station procedures to be used and does not have to be tested.

d. The monorail hoist does not have to be tested to move other than nuclear fuel. The surveillance test allows checks that do not pertain to the activity to be performed to be deferred (NA).

KA 234000 2.2.24 (3.8)
10CFR55.43(7)
New
Cognitive level: 2
Difficulty: 3

Provide to Candidate: T.S. 3.9.1

Enabling Objectives

COR0012102001040B Given a specific Refueling and Servicing Equipment malfunction, determine the effect on any of the following: Fuel handling operations

COR0012102001040C Given a specific Refueling and Servicing Equipment malfunction, determine the effect on any of the following: Core alterations

Skills

2.2.24 Ability to analyze the affect of maintenance activities on LCO status. (CFR: 43.2 / 45.13)

Question 100   765   (1 point(s))

A fuel asembly being transferred from the reactor core to the spent fuel pool - just entering the fuel pool from the Fuel Transfer Channel (cattle chute). The Area Radiation Monitor on the refueling bridge and adjacent to the spent fuel pool both alarm.

If it is desired to place the fuel assembly in a safe condition, which one of the following actions is required to place the fuel in the safest position BEFORE evacuting the refueling floor?

a. Place the fuel assembly into the nearest rack until seated.

b. Press the STOP pushbutton to deenergize the main hoist.

c. Confirm that the grapple ENGAGED indicating light is on.

d. Close the air supply valves for pneumatics to the main hoist.

Answer 100

a. Place the fuel assembly into the nearest rack until seated.

Reference: 0.24; 10.3

Foils:

b. Safest location is seated so the feul assembly cannot drop.
c. Safest location is seated so the feul assembly cannot drop.
d. Safest location is seated so the feul assembly cannot drop.

KA 2.2.28 (3.8)
10CFR55.43(7)
Bank
Cognitive Level: 1
Difficulty: 2

Enabling Objectives

COR0012102001030B   Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Refuel floor radiation levels/airborne levels

Skills

2.2.28 Knowledge of new and spent fuel movement procedures. (CFR: 43.6 / 45.12)