Proposed Question: RO Question # 1

While HPCI is in a CST to CST lineup for surveillance testing, the following occurs:

- Annunciator 1C06A (C-8) CST 1T-5A LO-LO LEVEL alarms
- Annunciator 1C06A (C-9) CST 1T-5B LO-LO LEVEL alarms
- Annunciator 1C03C (D-3) CST A/B LO LEVEL HPCI/RCIC SUCTION TRANSFER INITIATE alarms.
- MO-2321 INBD TORUS SUCTION ISOLATION and MO-2322 OUTBD TORUS SUCTION ISOLATION open.

Which one of the following is the correct system response?

MO-2300 CST SUCTION closes when __(1)__ MO-2321 and/or MO-2322 are full open.

The in service Condensate Service Water pump is tripped due to ___(2)___.

A. (1) both
(2) overcurrent

B. (1) either
(2) overcurrent

C. (1) both
(2) low CST level

D. (1) either
(2) low CST level

Explanation (Optional):

A. Incorrect - Both MO-2321 full open and MO-2322 full open will close the CST suction valve, while the Condensate Service Pump will trip at CST level of 6.25 ft in CST.

B. Incorrect - Both MO-2321 full open and MO-2322 full open will close the CST suction valve, while the Condensate Service Pump will trip at CST level of 6.25 ft in CST.

C. Correct - Both MO-2321 full open and MO-2322 full open will close the CST suction valve, while the Condensate Service Pump will trip at CST level of 6.25 ft in CST.

D. Incorrect - Both MO-2321 full open and MO-2322 full open will close the CST suction valve, while the Condensate Service Pump will trip at CST level of 6.25 ft in CST.

Technical Reference(s): OI-152, Section 8.1 & 8.2, pgs 31 & 32.
SD 152, pg 29
SD 537
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC RO Bank 19199

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 2

Which one of the following describes the relationship between the Standby Liquid Control System (SBLC) and the Core Spray (CS) line break detection system?

A differential pressure switch measures the pressure difference between the ____(1)_____ AND the inside of the ______(2)_______

A. (1) below core plate (inner pipe of the SBLC penetration)
(2) reactor pressure vessel in the downcomer annulus region.

B. (1) above core plate (outer pipe of the SBLC penetration)
(2) reactor pressure vessel in the downcomer annulus region.

C. (1) below core plate (inner pipe of the SBLC penetration)
(2) CS sparger pipe, just outside the reactor vessel.

D. (1) above core plate (outer pipe of the SBLC penetration)
(2) CS sparger pipe, just outside the reactor vessel.

Explanation (Optional):

A. Incorrect - A differential pressure switch measures the pressure difference between the bottom of the core which is the outer pipe of the SBLC penetration. The inside of the core spray sparger pipe measures the pressure inside the core shroud.

B. Incorrect - The inside of the core spray sparger pipe measures the pressure inside the core basket.

C. Incorrect - A differential pressure switch measures the pressure difference between the bottom of the core which is the outer pipe of the SBLC penetration.

D. Correct - A differential pressure switch measures the pressure difference between the bottom of the core which is the outer pipe of the SBLC penetration. The inside of the core spray sparger pipe measures the pressure inside the core shroud.

Technical Reference(s): SD 151, pgs 20 - 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question:	RO Question # 3

The plant is operating in MODE 1 at 100% power with the following conditions:

- The Startup Transformer is removed from service due to preplanned maintenance
- The Standby Transformer is powering busses 1A3 and 1A4
- A LOCA occurs
- RPV level lowered to 30 inches before recovering to 175 inches

What is the response to this event of the RBCCW Drywell Supply and Return Isolation Valves, MO-4841A and MO-4841B?

MO-4814A and MO-4841B will ________.

A. remain OPEN (no isolation associated)

B. NOT go closed automatically due to loss of power to both isolation valves

C. go closed and will automatically reopen with no other operator actions

D. go closed and will require operator actions to reset the isolation and open the valves

Explanation (Optional):

A. Incorrect – Valves will close on Group 7 isolation

B. Incorrect – Valves are powered by 1B42 (essential power)

C. Incorrect – Group 7 required to be reset on 1C31 prior to opening valves.

D. Correct - The valve solenoids for the Drywell Cooling Isolation Valves (CV) are powered by 120 VAC Instrument AC from 1Y11 and 1Y21, and are Energize-to-Close. The Motor-Operated valves for RBCCW are powered from 480 VAC 1B42.
For Group 7, the RBCCW and Well Water Isolations Seal In with the use of the Aux Relay, CR-4841X. When the Reactor Low-Low-Low Level Sensor Relays reset, the Reset pushbutton on 1C31 will need to be depressed to reset the Group 7 Isolation signal. There is an amber indicating light at 1C31 to indicate when the Isolation Signal is Locked In. When the Isolation Signal is Reset, then the Drywell Cooling solenoid valves will reopen automatically if Drywell Cooling is on, but the motor-operated valves for RBCCW will need to be reopened.

Technical Reference(s): SD 414, pg 10
SD 959-1, pg 40, 43 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 4

The plant conditions are as follows:

- Backup Instrument Air Compressor 1K1 is in the STANDBY-operating mode
- 1K1 electrical power is being supplied from 480 VAC Bus 1B33

A large electrical disturbance occurs resulting in:
- LLRPSF transformers XR1 and XR2 de-energizing, and
- A Bus 1A3 lockout.

Which one of the following describes the response of the Backup Instrument Air Compressor 1K1?

1K1 will ______.

A. need to have it's power supply is transferred from 1B33 to 1B45 to start

B. start when header pressure reaches 100 psig and will cycle to maintain 100 - 110 psig

C. start when header pressure reaches 90 psig and will cycle to maintain 90 - 100 psig

D. need to have HSS-3002, BACKUP COMPRESSOR 1K-1 PRESSURE SELECT SWITCH placed in the PRIMARY position to start

Explanation (Optional):

A. Correct - The transfer switch 1N3312 is a "break before make" type which is operated to allow 1K-1 to be powered from either 1B3312 or 1B4501. 1B3312 will be the normal power supply selected. If 1B33[1B45] has to be de-energized for any reason, the compressor power can be transferred to 1B4501[1B3312].

B. Incorrect - This condition is not a trip, but without power the compressor does not run.

C. Incorrect - This condition is not a trip, but without power the compressor does not run.

D. Incorrect - This condition is not a trip, but without power the compressor does not run.

Technical Reference(s): OI-518.1, Sect 4.7, pg 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC RO 19111

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 5

The plant is operating in MODE 1 at 93% power with the following plant conditions:

- “A” and “B” APRM’s are bypassed to support LPRM whisker burns.
- LPRM 4D-08-09, an LPRM shared between APRM “A” and “B” fails upscale.

Which of the following describes the affect of this failure on the value of computer point C179, NSSS1 CORE THERMAL POWER (MWTH)?

A. "B" APRM reading will increase causing C179 to RISE

B. "B" APRM reading will increase, however, since the APRM is bypassed C179 will REMAIN THE SAME

C. LPRMs do NOT input into the Reactor Heat Balance Equation and therefore C179 will REMAIN THE SAME

D. "B" APRM readings will lower because the "D" Level LPRM upscale reading is automatically rejected causing C179 to LOWER

Explanation (Optional):

A. Incorrect - The affect on the B APRM is correct but it has no effect on the heat balance

B. Incorrect - The affect on the B APRM is correct but it has no effect on the heat balance

C. Correct - The heat balance is used to adjust APRM gains, LPRMs and APRMs are not inputs to MWTH

D. Incorrect - LPRMs are not automatically rejected in APRMs, however in the RBM system they are.

Technical Reference(s): SD-878.3, Rev 8; Pages 44-45. SD-900, Rev. 4, pgs. 7-9. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 81.01.01.15 (As available)

Question Source: Bank # 2005 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2

55.43

General Design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Proposed Question: RO Question # 6

Plant conditions are as follows:

- Manual and automatic actions have failed to insert control rods
- RPV Flooding EOP has been entered

How is adequate core cooling assured during this event?

Depressurize the RPV, then control injection to establish and maintain ___(1)___. The core will then be cooled by ___(2)___.

A. (1) RPV level between -25 in. and +211
(2) submergence or Steam Cooling

B. (1) RPV level between -25 in. and the level required to lower power below 5%
(2) full submergence

C. (1) RPV pressure above the Minimum Steam Cooling Pressure
(2) submergence or Steam Cooling

D. (1) RPV level flooded to the elevation of the RPV flange and RPV pressure a minimum of 150 psig above Torus pressure
(2) Steam Cooling

Explanation (Optional):

A. Incorrect - This is the broad range of water level requirements during an ATWS it would not apply if RPV Flooding is entered.

B. Incorrect - This is the broad range of water level requirements during an ATWS it would not apply if RPV Flooding is entered.

C. Correct - RPV flooding, is used to cool the core when RPV water level cannot be determined. The specified actions first depressurize the RPV, then control injection to establish and maintain one of the following conditions:

- The RPV flooded to the elevation of the main steam lines. The core will then be cooled by full submergence. This condition may ultimately be achieved under either shutdown or failure-to-scram conditions.
- RPV pressure above the Minimum Steam Cooling Pressure. The core will then be cooled by submergence or steam cooling. Since reactor power must be at least 6%-10% to generate the amount of steam required to sustain the Minimum Steam Cooling Pressure, this condition is applicable only under ATWS conditions.

D. Incorrect – The direction of RPV/F EOP is to maintain water level at the Main Steam Lines, not the RPV head. The 150 psig is the minimum steam cooling pressure for 4 SRVs open.

Technical Reference(s): EOP RPV Flooding Bases pg 2
RPV Flooding step F-7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 7

During a manual start of RCIC, the following indications are observed:

- TURBINE STEAM SUPPLY MO-2404 starts to open
- RCIC Turbine speed begins to rise
- RCIC Pump Discharge pressure begins to rise

At this point annunciator 1C04C, A-5, RCIC MO-2405 TURB TRIP alarms followed 5 seconds later, by the following alarms:

- Annunciator 1C04C, D-9, RCIC TURBINE BEARING OIL LO PRESSURE alarms.
- Annunciator 1C04C, B-4, RCIC LO FLOW alarms.
- Reactor water level is 186 inches and stable.

No other alarms are present on 1C04C and all alarms are in proper working order.

Which one of the following provides the correct analysis of this situation?

A. A turbine trip has occurred on low flow.

B. A turbine trip has occurred on overspeed.

C. A turbine trip has occurred on low oil pressure.

D. A turbine trip has occurred but MO-2404, TURBINE STEAM SUPPLY, has failed to automatically close.

Explanation (Optional):

A. Incorrect - A low flow condition would not cause a turbine trip

B. Correct - A turbine overspeed trip will only cause an alarm on the turbine trip. The low oil pressure and low flow result from the turbine speed coasting down after the RCIC turbine trip.

C. Incorrect - During a loss of oil pressure the turbine will overspeed because the RCIC turbine control valve is opened by spring pressure and closed by oil pressure.

D. Incorrect - There is no indication that MO-2404 failed to close.

Technical Reference(s): OI-150, pg 48 and SD -150 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 8

The plant is operating in MODE 4 in Shutdown Cooling with the following conditions:

- RPV level is 190 inches
- The “A” RHR pump and “A” RHRSW pump are running

An event occurs that causes RPV level to rapidly drop to 50 inches.

Which one of the following describes how the RHR pumps automatically respond to the signal?

	A RHR Pump		C RHR Pump	
A.	Remains in Shutdown Cooling		Starts and operates on minimum flow	
B.	Trips and does not restart		Starts and operates on minimum flow	
C.	Trips and does not restart		Attempts to start and immediately trips	
D.	Remains in Shutdown Cooling		Attempts to start and immediately trips	

Explanation (Optional):

A. Incorrect - The "A" pump trips. All others start but C trips

B. Incorrect - The "A" pump trips and does not restart. All others start but C trips

C. Correct - Per SD 149, page 22 - In the event a LOCA occurs when the RHR System is in the shutdown cooling mode, the RHR System will not automatically realign itself for LPCI injection. Operator actions required to initiate the LPCI mode of RHR include resetting the Group 4 Isolation Seal-In, restoring torus suction flowpath to the RHR pumps, and manually restarting the RHR pumps that have tripped.
Additionally, the SDC suction valves close on the LPCI signal (PCIS Group 4). The "C" RHR Pump breaker will receive a start signal but immediately trip. The trip occurs due to no suction path present to prevent pump damage. This is NOT a start permissive, it s a pump trip (SD-149 page 12)

D. Incorrect - The "A" pump trips. The "C" RHR Pump breaker will receive a start signal but immediately trip. The trip occurs due to no suction path present to prevent pump damage. This is NOT a start permissive, it s a pump trip (SD-149 page 12).

Technical Reference(s): SD 149 Rev 11 pages 12 & 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 2005 NRC Exam
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2005

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 9

The plant was operating in MODE 2 at 7% power when an accident occurred. Current plant conditions are as follows:

- DW pressure is 8 psig, rising
- HPCI system tripped
- RCIC injecting into RPV @ 415 gpm
- RPV level reaches 64 inches and lowering at Time Zero (T0)

Assuming no operator action, the ADS system will automatically actuate to lower RPV pressure when any lower pressure ECCS pump ___(1)___ and with ___(2)___ (referenced to time zero).

A. (1) breaker is CLOSED
(2) no time delay

B. (1) breaker is CLOSED
(2) a two minutes time delay

C. (1) reaches normal discharge pressure
(2) no time delay

D. (1) reaches normal discharge pressure
(2) a two minutes time delay

Explanation (Optional):

A. Breaker closed is not the correct signal; triple low level must be in place two minutes

B. Breaker closed is not the correct signal

C. ADS waits two minutes

D. Timer starts when reactor water level reaches low-low-low level. Two minutes later, if an RHR or Core Spray pump is at normal discharge pressure, ADS will open 4 SRVs. This assumes that timers are not overridden

Technical Reference(s): SD-183.1 Rev. 6, Page 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 8.02.01.02 (As available)

Question Source: Bank # 2005 NRC

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Proposed Question: RO Question # 10

A plant shutdown is in progress and conditions are as follows:

- Reactor is in MODE 3 with RPV pressure 30 psig
- Reactor water level is 190 inches on all GEMAC level instruments
- Both reactor recirculation pumps are shutdown
- A loss of Shutdown Cooling occurs and RHR CANNOT be recovered

Which one of the following would provide an alternate method to ensure core DECAY HEAT REMOVAL is re-established?
(Assume no Defeats are installed.)

A. Start RCIC in CST-To-CST mode to lower RPV pressure

B. Raise RPV level to +214 inches using HPCI to provide natural circulation

C. Starting one of the Reactor Recirculation pumps to re-establish recirculation flow

D. Raise RPV level with a Condensate pump and perform feed and bleed to the torus with SRVs

Explanation (Optional):

A. Incorrect - Reactor pressure is below RCIC isolation setpoint making RCIC unavailable.

B. Incorrect - Reactor pressure is below HPCI isolation setpoint making HPCI unavailable.

C. Incorrect - Because the restoration of a Recirc pump does not result in heat removal.

D. Correct - Because these actions are consistent with guidance in AOP-149, Inadequate Decay Heat Removal.

Technical Reference(s): AOP-149, Sect 4.2, pg 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 11421

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 11

The plant was operating in MODE 1 at 44% power with the following conditions:

- HPCI was inoperable for preplanned maintenance

A LOCA then occurred resulting in the following plant conditions:

- DW Pressure is 7 psig rising slowly
- All control rods fully inserted
- RPV Pressure is 730 psig, lowering slowly
- RPV Level is 60 inches, lowering slowly
- ADS timers initiated and are timing out
- With 30 seconds left on the ADS timers, the "A" ADS timer loses power

Which one of the following describes the status the ADS Valves and Core Spray Pump(s) when the B ADS logic times out?

ADS Valves ___(1)___
Core Spray Pump(s) _____(2)_____

A. (1) will remain closed.
(2) "B" ONLY remains on minimum flow.

B. (1) 4400 and 4405 only will open.
(2) "A" and "B" inject when pressure lowers below their discharge head.

C. (1) 4400, 4402, 4405 and 4406 will open.
(2) "B" ONLY injects when pressure lowers below its discharge head.

D. (1) 4400, 4402, 4405 and 4406 will open.
(2) "A" and "B" inject when pressure lowers below their discharge head.

Explanation (Optional):

A. Incorrect - Although the channel A timer is not energized, ADS will actuate with only one channel timed out.

B. Incorrect - Only one timer needs to time out to actuate the all ADS valves.

C. Incorrect - The loss of power to the ADS logic does not affect the Core Spray pumps since this logic is not shared, both pumps will inject.

D. Correct Only one timer needs to time out to actuate the all ADS valves and the ADS logic does not affect the Core Spray pumps, both pumps will inject.

Technical Reference(s): SD-183.1, pg 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 12

The plant is operating in MODE 1 at 100% power with the following plant conditions:

- The “B” SBDG is tagged out for heat exchanger replacement.
- A tornado strikes the switchyard causing a loss of off-site power (LOOP).

Assuming no operator action, which one of the following is the status of the Standby Gas Treatment (SBGT) systems?

A. Both SBGT trains remain in STANDBY and are available to start on an initiation signal

B. ONLY the "A" SBGT has received a start signal and it has automatically started

C. Both SBGT lockout relays tripped but only the “A” SBGT train is running

D. Both SBGT lockout relays have tripped and both SBGT trains are running

Explanation (Optional):

A. Incorrect - The loss of off-site power results in a loss of RPS which causes an initiation of both SBGT systems. The 480V Bus 1B34 will be supplied by the "A" Diesel which will allow an auto start of SBGT "A".

B. Incorrect - The loss of off-site power results in a loss of RPS which causes an initiation of both SBGT systems. The 480V Bus 1B34 will be supplied by the "A" Diesel which will allow an auto start of SBGT "A".

C. Correct - The loss of off-site power results in a loss of RPS which causes an initiation of both SBGT systems. The 480V Bus 1B34 will be supplied by the "A" Diesel which will allow an auto start of SBGT "A".

D. Incorrect - The loss of off-site power results in a loss of RPS which causes an initiation of both SBGT systems. The 480V Bus 1B34 will be supplied by the "A" Diesel which will allow an auto start of SBGT "A".

Technical Reference(s): AOP-358, ARP-1C05B (C-8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 13

A reactor startup from a 6 day maintenance outage is in progress. The reactor is in MODE 2 and control rod withdrawal is in progress with power in the IRM range.

As power rises, the IRM range switches shall be moved to maintain the IRM indication between ___(1)___ on the ___(2)___ scale and between ___(3)___ on the ___(4)___ scale.

A. (1) 3/40 and 25/40
(2) Odd
(3) 10/125 and 75/125
(4) Even

B. (1) 10/125 and 75/125
(2) Odd
(3) 3/40 and 25/40
(4) Even

C. (1) 10/40 and 25/40
(2) Odd
(3) 25/125 and 100/125
(4) Even

D. (1) 25/125 and 100/125
(2) Odd
(3) 10/40 and 25/40
(4) Even

Explanation (Optional):

A. Correct - IAW OI-878.2, Continue to reposition the IRM range switches to maintain indications on the IRM recorders between 10/125 and 75/125 on the Even scale and between 3/40 and 25/40 on the Odd scale..

B. Incorrect - Indication should be between 10/125 and 75/125 on the Even scale and between 3/40 and 25/40 on the Odd scale.

C. Incorrect - Indication should be between 10/125 and 75/125 on the Even scale and between 3/40 and 25/40 on the Odd scale.

D. Incorrect - Indication should be between 10/125 and 75/125 on the Even scale and between 3/40 and 25/40 on the Odd scale.

Technical Reference(s): OI-878.2, pg 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 14

The plant is operating in MODE 1 at 100% power with the following plant conditions:

- The “B” RPS MG set is to be secured to support planned maintenance
- The RPS Half Scram Preparation checklist is in progress
- The CRS directs that Reactor Water Cleanup be secured

Which one of the following actions must be performed in accordance with OI 261, Reactor Water Cleanup System?

A. Substitute RWCU System Flow computer point (B017) to indicate zero to maintain an accurate heat balance.

B. Open MO-2732, "RWCU Drain to Radwaste", to ensure the system depressurizes completely while it is isolated.

C. Inform Chemistry that the RWCU system is isolated and to commence taking manual RWCU system grab samples.

D. Isolate the Non-Regenerative Heat Exchanger by isolating the shell side RBCCW flow before isolating the tube side RWCU flow.

Explanation (Optional):

A. Correct - IAW OI-261, Computer point B017, RWCU System Flow, may need to be substituted to zero, during system shutdown/isolation, to maintain accurate 3D Monicore periodic logs.

B. Incorrect - There is no need to drain the system.

C. Incorrect - Manual grab samples would be required if the system was operating and the normal sampling system was not operable.

D. Incorrect - The entire system is to be isolated not the Non-Regenerative Heat Exchanger.

Technical Reference(s): OI-261, pg 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Sys ID 18933

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 15

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

- A spurious Group 1 isolation occurs
- Low Low Set (LLS) SRVs actuate to control pressure
- One LLS SRV tailpipe vacuum breaker is stuck open such that Containment pressure is 1.2 psig and rising slowly

(1) What is the result of this condition? AND
(2) What actions need to be taken?

A. (1) Steam from the SRV will go into the Drywell atmosphere
(2) Install EOP Defeat 9 and vent the drywell via SBGT.

B. (1) Steam from the SRV will go into the Drywell atmosphere
(2) AOP 573 may be used to vent the drywell via SBGT as long as containment pressure is < 2.0 psig.

C. (1) Steam from the SRV will go into the Torus atmosphere
(2) Install EOP Defeat 9 and vent the drywell via SBGT.

D. (1) Steam from the SRV will go into the Torus atmosphere
(2). AOP 573 may be used to vent the drywell via SBGT as long as containment pressure is < 2.0 psig.

Explanation (Optional):

A. Incorrect - The SRV vacuum breaker being open allows direct communication of some steam to the DW air space NOT the Torus airspace. Defeat 9, High Drywell Pressure and RPV low level defeat is not authorized in this situation.

B. Correct - The SRV vacuum breaker being open allows direct communication of some steam to the DW air space that may raise DW pressure. AOP-573 directs venting the DW if pressure rises to 1.0 to 1.5 psig by venting Drywell through SBGT.

C. Incorrect - The SRV vacuum breaker being open allows direct communication of some steam to the DW air space NOT the Torus airspace. Defeat 9, High Drywell Pressure and RPV low level defeat is not authorized in this situation.

D. Incorrect - The SRV vacuum breaker being open allows direct communication of some steam to the DW air space that may raise DW pressure. AOP-573 directs venting the DW if pressure rises to 1.0 to 1.5 psig by venting Drywell through SBGT.

Technical Reference(s): AOP-573
SD 183-1, pg 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 16

The plant is starting up in Mode 1 at 12% power with the following conditions:

- A RFP is in service
- The Startup Feedwater Control Valve CV-1622 is in service in Auto

Which one of the following describes how a loss of Instrument Air will affect CV-1622 and what actions are required to control Reactor water level?

Feedwater Startup Control Valve CV-1622 fails ___(1)___ .
Control Reactor water level by ___(2)___ IAW AOP 644, FEEDWATER/ CONDENSATE MALFUNCTION.

A. (1) open
(2) throttling the Startup Feedline Block Valve MO-1631 CLOSED

B. (1) closed
(2) OPENING Feed Regulating Valve CV-1579 as appropriate

C. (1) locked up (as-is)
(2) tripping feedwater pumps or CLOSING Feed Regulating Valve CV-1579 as appropriate.

D. (1) locked up (as-is)
(2) throttling the Startup Feedline Block Valve MO-1631

Explanation (Optional):

A. Incorrect - With a loss of air the Startup Feed Reg Valve will lock up (fail as-is). If the failure lasts longer than 30 minutes, the FRV will tend to drift open (even locked up).

B. Incorrect - With a loss of air the Startup Feed Reg Valve will lock up (fail as-is). If the failure lasts longer than 30 minutes, the FRV will tend to drift open (even locked up).

C. Incorrect - With a loss of air the Startup Feed Reg Valve will lock up (fail as-is). If the failure lasts longer than 30 minutes, the FRV will tend to drift open (even locked up). There is no direction to trip the feedwater pumps to maintain Reactor water level.

D. Correct - With a loss of air the Startup Feed Reg Valve will lock up (fail as-is). If the failure lasts longer than 30 minutes, the FRV will tend to drift open (even locked up). ARP-1C05A, E-1 directs throttling Blocking Valve MO-1631 or opening Feed Reg Valve CV-1579(1621) as appropriate.

Technical Reference(s): ARP-1C05A, E-1
AOP 518, page 5 Note & step 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 17

The plant is starting up in MODE 3 with the following conditions:

- Reactor Pressure at 675 psig
- Both ESW pumps were operating to support torus cooling operations.
- A loss of offsite power (LOOP) occurs with all systems operating as designed.

Which one of the following correctly states:
(1) When will the ESW pumps restart?
(2) What is the ESW flowrate compared to prior to the loss of offsite power (more or less)?

A. (1) when the SBDGs are supplying the bus
(2) less

B. (1) when the SBDGs are supplying the bus
(2) more

C. (1) Pumps will NOT auto start
(2) less

D. (1) Pumps will NOT auto start
(2) more

Explanation (Optional):

A. Incorrect - ESW flow will be greater than before the LOOP because the cooling water valves opened for Torus cooling will remain open while the ESW flow to the emergency diesel generators will open under control of the SBDG start logic.

B. Correct - The ESW pumps start automatically if the associated emergency diesel generator starts. ESW flow will be greater than before the LOOP because the cooling water valves opened for Torus cooling will remain open while the ESW flow to the emergency diesel generators will open under control of the SBDG start logic.

C. Incorrect - The ESW pumps start automatically if the associated emergency diesel generator starts. ESW flow will be greater than before the LOOP because the cooling water valves opened for Torus cooling will remain open while the ESW flow to the emergency diesel generators will open under control of the SBDG start logic.

D. Incorrect - The ESW pumps start automatically if the associated emergency diesel generator starts.

Technical Reference(s): SD-454, pg 7 & 8. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 18

The plant is operating in MODE 1 at 100% power with the following conditions:

- The 1Y23 Power Source Manual Transfer Switch (HSS-1Y23A) is in the AUTO TO 1Y2 position
- The voltage at 1Y23 lowers to 100 VAC and then recovers to 120 VAC

Which ONE of the following describes the affect of this transient on Uninterruptible Power System loads?

Loads will be …

A. continuously powered from 1D45/1Y4.

B. interrupted by a momentary BREAK BEFORE MAKE transfer to 1Y2 and remain powered from 1Y2.

C. continuously powered during the MAKE BEFORE BREAK transfer to 1Y2 and then automatically transfer back to 1D45/1Y4 when voltage recovers.

D. interrupted by a momentary BREAK BEFORE MAKE transfer to 1Y2 and then automatically transfer back to 1D45/1Y4 when voltage recovers.

Explanation (Optional):

A. Incorrect – This would be true if voltage lowered to 115 VAC and recovered.

B. Correct - When voltage lowers to 105 VAC, device 27-22 forces a break before make transfer to 1Y2. Operator action is required to enable transfer back to 1D45/1Y4.

C. Incorrect – This would be true if 1Y22 operated like the Static Switch.

D. Incorrect – This would be true if 1Y23 Power Source Manual Transfer Switch (HSS-1Y23A) were in the 1D45/1Y4 position.

Technical Reference(s): SD-357 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 2007 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 19

The plant was operating in MODE 1 at 100% power with the following conditions:

- A severe electrical transient has occurred resulting in a station blackout
- AOP 301.1, Station Blackout, has been entered
- The grid operator reports that power has been restored to the DAEC switchyard
- Normal voltage conditions are expected to be restored within the next 30 minutes

The BOP reports the following from 1C08:

- The GENERATOR OUTPUT H BREAKER Synchronizing Switch is ON
- The RUNNING voltmeter reads 82 volts

Can the Essential Buses 1A3 and 1A4 be restored using the Standby Transformer until normal voltage is restored to the grid?

A. No, the Degraded Voltage Relays cannot be reset

B. Yes, provided the Degraded Voltage Relays are reset at 1C08 only.

C. No, the Degraded Voltage Relays cannot be reset at 1C08 and then overridden at 1C351/1C352.

D. Yes, the Degraded Voltage Relays must be reset at 1C08 and then overridden at 1C351/1C352.

Explanation (Optional):

A. Incorrect - The low voltage can be overridden.

B. Incorrect - The degraded voltage can NOT be reset at this voltage, voltage must be above 96% (111 volts) to reset.

C. Incorrect - The low voltage can be overridden.

D. Correct – Overriding the degraded voltage will work if incoming voltage is more than 65% (2700 Volts) (incoming of 78 volts). If degraded grid voltages exist, override degraded bus voltage condition on essential buses 1A3/1A4 by resetting the degraded voltage relays at 1C08 by pushing the degraded voltage reset pushbuttons, then override the Degraded Voltage Relays at 1C351[1C352] using TEST switches.

Technical Reference(s): AOP-301.1, pg 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank #19551

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question:	RO Question # 20

The reactor is in MODE 2 with a reactor startup in progress with the following conditions:

- No SRM's or IRM's are bypassed
- The SRM detectors are being withdrawn per IPOI-2, Startup

Which one of the following sets of conditions will result in activation of alarm 1C05A (E-5), SRM DETECTOR RETRACTED WHEN NOT PERMITTED?

	All IRM Range Switch Positions	A SRM Reading	B SRM Reading	C SRM Reading	D SRM Reading
A.	1	120 cps	120 cps	120 cps	120 cps
B.	2	90 cps	150 cps	150 cps	150 cps
C.	3	90 cps	90 cps	90 cps	90 cps
D.	4	90 cps	120 cps	120 cps	120 cps

Explanation (Optional):

A. Incorrect - plausible; would be true if SRM counts were given below 100 cps

B. Correct - With detectors partially withdrawn, an SRM reading 90 cps will generate SRM DETECTOR RETRACTED WHEN NOT PERMITTED alarm with IRMs on range 2.

C. Incorrect - plausible; would be true if IRMs were given below range 3

D. Incorrect - plausible; would be true if IRMs were given below range 3

Technical Reference(s): ARP 1C05A E-5 Rev 58 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 11263

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 21

The plant is in MODE 4 with the following conditions:

- Refueling is in progress
- It becomes necessary to remove a 125 VDC Station Battery from service

Which one of the following is the Technical Specifications implication of removing this battery from service?

The affected 125 VDC Power DISTRIBUTION System ...

A. shall be considered inoperable and the appropriate LCO entered.

B. is operable provided its associated battery charger is operable.

C. is operable provided two independent battery chargers are operable.

D. shall be considered inoperable but is not required in this plant condition.

Explanation (Optional):

A. Correct - If a battery is disconnected and only a charger is supplying the bus; the affected 125 VDC Power Distribution System shall be considered inoperable. With a required 125 VDC battery or distribution subsystems inoperable during SDC operations, Core Alts, OPDRVs, moving fuel, etc, either immediately declare inoperable any required features that are dependent on 125 vdc, or immediately suspend all such activities.

B. Incorrect - If a battery is disconnected and only a charger is supplying the bus; the affected 125 VDC Power Distribution System shall be considered inoperable.

C. Incorrect - If a battery is disconnected and only a charger is supplying the bus; the affected 125 VDC Power Distribution System shall be considered inoperable.

D. Incorrect - With a required 125 VDC battery or distribution subsystems inoperable during SDC operations, Core Alts, OPDRVs, moving fuel, etc, either immediately declare inoperable any required features that are dependent on 125 VDC, or immediately suspend all such activities.

Technical Reference(s): OI-302, pgs 4 & 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

12-29-10-is this OK for ROs

Proposed Question: RO Question # 22

The plant is SHUTDOWN in MODE 4 for a maintenance outage with the following conditions:

- All APRMs are currently OPERABLE
- The “A” and “D” APRM’s are currently bypassed

Due to a maintenance activity, the CRS directs the “C” APRM be bypassed.

What other APRM, if any, shall be bypassed IAW approved procedures?

A. APRM “B” shall be bypassed using the APRM bypass switch on the LEFT side of 1C05.

B. APRM “B” shall be bypassed using the APRM bypass switch on the RIGHT side of 1C05.

C. APRM “D” shall remain bypassed, can be verified using the APRM bypass switch on the LEFT side of 1C05.

D. APRM “D” shall remain bypassed, can be verified using the APRM bypass switch on the RIGHT side of 1C05.

Explanation (Optional):

A. Incorrect - With C bypassed, the companion APRM that should be bypassed is “B” APRM. The “B” APRM is bypassed using APRM bypass switch on the right side of 1C05.

B. Correct - With C bypassed, the companion APRM that should be bypassed is “B” APRM. The “B” APRM is bypassed using APRM bypass switch on the right side of 1C05.

C. Incorrect - With C bypassed, the companion APRM that should be bypassed is “B” APRM. The “B” APRM is bypassed using APRM bypass switch on the right side of 1C05.

D. Incorrect - With C bypassed, the companion APRM that should be bypassed is “B” APRM. The “B” APRM is bypassed using APRM bypass switch on the right side of 1C05.

Technical Reference(s): OI-878.4, P&L 12, NOTE on p11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 23

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

- Drywell pressure is currently 10 psig, rising slowly
- ADS has initiated and all 4 ADS valves are open
- RHR Pumps A and C are running on minimum flow
- Both CS pumps will not start
- RHR Pumps B and D will not start
- RPV pressure is 750 psig, lowering
- RPV level is 32 inches, lowering

Which one of the following conditions would cause the ADS valves to close?

A. Securing either RHR Pump.

B. Raising RPV level to 65 inches

C. Securing both the RHR Pumps

D. Reducing RPV pressure to 100 psig

Explanation (Optional):

A. Incorrect - Either RHR Pump running will provide a permissive for ADS valves to remain open.

B. Incorrect - Clearing the Low Level setpoint will NOT close the SRVs because after the system initiates this signal is bypassed.

C. Correct - Securing both RHR Pumps removes the permissive for the SRVs to open causing them to close.

D. Incorrect - The SRVs will remain open until reactor system pressure lowers to approximately 50 psi above Drywell/Torus pressure, the pilot valve will reseat and the main valve spring pressure will reseat the main disc. In this case approximately 60 psig.

Technical Reference(s): SD-183-1, pg 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	DAEC #19343	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 24

The plant is in MODE 5 with Refueling currently in progress. Mode Switch is in REFUEL.

Which one of the following would result in a FULL reactor scram?

A. CRD Scram Discharge Volume high level trip of 60 gallons

B. Inadvertent closure of all of the OUTBOARD MSIVs

C. Intermediate Range Monitor "A" upscale spike to 120/125 on Range 1 due to undervessel work.

D. Tripping of the Main Turbine at 1C07 using the Turbine Trip pushbutton

Explanation (Optional):

A. Correct - CRD Scram Discharge Volume High Water Level is sensed in the instrument volume. A level of 60 gallons will result in a full reactor scram.

B. Incorrect - With the plant shutdown for refueling the MSIV isolation scram is bypassed.

C. Incorrect - A single IRM trip would only cause a half scram.

D. Incorrect - With the plant shutdown for refueling the turbine stop valve scram is bypassed.

Technical Reference(s): SD-358, pg 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 25

The plant is in MODE 5 with RPV level at the RPV flange in preparation for flood up. Core Spray keylock switch E21A-S16A SUCTION PATH INTERLOCK HS-2103A is placed in the BYPASS position.

What is the bases for placing the switch in the BYPASS position?

This switch…

A. overrides the automatic opening of the Core Spray suction valves on a system initiation.

B. permits closing the Core Spray suction valve when the CST suction valve is opened.

C. overrides the automatic opening of the Core Spray minimum flow valve when a CST suction valve is open.

D. permits the pump to be run with suction from the condensate storage tanks, with the torus suction path isolated.

Explanation (Optional):

A. Incorrect - The switch has no function related to an automatic initiation.

B. Incorrect - The valves can be repositioned prior to placing the switch in bypass.

C. Incorrect - The switch has no function related to the minimum flow valve.

D. Correct - In order to provide for use of the condensate storage tanks as an alternate suction source, keylocked Core Spray Pump A [B] Suction Path Intlk switches on panel 1C43 [1C44] bypass the loss of suction path interlock when placed in BYPASS. This permits the pumps to be run with suction from the condensate storage tanks, with the torus suction path isolated.

Technical Reference(s): OI-151, Sect. 10, pg 31
SD-151, pgs 9 & 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 26

The plant is operating at 100% power when a loss of 120 VAC Instrument Bus 1Y21 occurs.

Which of the following describes the effect of this power loss on the RHR pumps?

A. On the power loss, ONLY RHR Pumps B and D automatically start and operate on minimum flow

B. On the power loss, all RHR Pumps automatically start

C. If a LPCI initiation signal is received, ONLY “A” and “C” RHR pumps would AUTO start

D. If a LPCI initiation signal is received, all RHR pumps would AUTO start as designed

Explanation (Optional):

A. Incorrect - No pump starts occur.

B. Incorrect - No pump starts occur.

C. Incorrect - RHR logics are cross-divisionalized such that a loss of one 120 VAC Instrument supply does not impact LPCI pump starts.

D. Correct - RHR logics are cross-divisionalized such that a loss of one 120 VAC Instrument supply does not impact LPCI pump starts.

Technical Reference(s): SD-317-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 27

The plant is operating in MODE 1 at 100% power with the following conditions:

- Repairs on "A" Rod Block Monitor have just been completed
- RBM A is removed from BYPASS to accomplish Post Maintenance Testing
- The ROD OUT PERMISSIVE light extinguished and then illuminated again within two seconds
- Annunciator 1C05B (A-6), ROD OUT BLOCK did NOT alarm

Which one of the following statements describes the system response to the above?

This condition is ...

A. normal because the annunciator has a 10 second time delay.

B. normal because "A" RBM generated a rod out inhibit during the null sequence.

C. NOT normal only because the annunciator should have alarmed when the ROD OUT PERMISSIVE light was extinguished.

D. normal because the rod out blocks are bypassed for two seconds to allow the reference APRM gain adjustment during the null sequence.

Explanation (Optional):

A. Incorrect - There is no delay on the annunciator, the RBM trip functions are bypassed during the nulling sequence so no alarm is generated.

B. Correct - Taking a RBM out of BYPASS initiates a null sequence. RBM trip functions are bypassed during the nulling sequence so no alarm is generated.

C. Incorrect - The RBM trip functions are bypassed during the nulling sequence so no alarm is generated.

D. Incorrect - There is no rod block bypass, the RBM trip functions are bypassed during the nulling sequence so no alarm is generated.

Technical Reference(s): SD-878-5, pg 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # LOT Bank 19363

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 28

With the plant operating at full power, the following alarms are received:

- 1C08B A-9, BUS 1A2 LOCKOUT TRIP OR LOSS OF VOLTAGE
- 1C06A D-12, CONDENSATE PUMPS 1P-8A/B LO DISCH PRESSURE
- 1C06A C-12, A RX FEED PUMP 1P-1A LOW SUCTION PRESS
- 1C06A C-13, B RX FEED PUMP 1P-1B LOW SUCTION PRESS

Which one of the following describes the status of operating Condensate and Feedwater Pumps?

A. ONLY the “A” Condensate Pump is operating.

B. ONLY the “B” Condensate Pump is operating.

C. The “A” Condensate Pump AND the “A” Feed Water Pump are operating.

D. The “B” Condensate Pump AND the “B” Feed Water Pump are operating.

Explanation (Optional):

A. Incorrect - Identifies potential misconception of 1P-1A Low Suction Pressure TRIP.

B. Incorrect - Would be true for Bus 1A1 Lockout with potential misconception of 1P-1A Low Suction Pressure TRIP.

C. Correct - Bus 1A2 Lockout de-energizes BOTH Condensate Pump 1P-8B AND Feed Water Pump 1P-1B.

D. Incorrect - Would be true for Bus 1A1 Lockout.

Technical Reference(s): SD-639 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 2007 NRC exam

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Proposed Question: RO Question # 29

The plant is operating in MODE 1 at 100% power with the following conditions:

- All LCO's are met

Which one of the following is a consequence of prolonged operation the Control Building Ventilation System in the PURGE mode?

The PURGE mode ...

A. bypasses the heating and cooling coils resulting in loss of Control Building temperature control.

B. isolates the outside air intake lowering Control Building pressure below atmospheric pressure.

C. ventilation flow bypasses the Cable Spreading and Battery Rooms which may result in having to declare the Batteries inoperable.

D. closes the Control Room Recirculation Damper which could result in more rapid buildup of radiological or toxic chemical concentrations.

Explanation (Optional):

A. Incorrect - When HS 6107 is placed in the Fresh Air mode of operation, the Control Room Recirculation Damper DO6109 fully closes, this mode does not bypass the heating and cooling and temperature is not a concern.

B. Incorrect - Damper Operator DO6106A(B) maintains mixing plenum (supply fan suction) .25"wg greater than outside pressure.

C. Incorrect - Placing the Control Building Ventilation system in the PURGE mode does not bypass the Cable Spreading and Battery Rooms.

D. Correct - When HS 6107 is placed in the Fresh Air mode of operation, the Control Room Recirculation Damper DO6109 fully closes. The basis for use of the fresh/auto (purge) mode is at the discretion of the OSM/CRS for comfort in the control room only. If the control building ventilation is operated in purge mode for extended periods, and a radiological or toxic chemical event were to occur, the higher intake flow rate in PURGE mode could result in more rapid buildup of radiological or toxic chemical concentrations than has been assumed in the safety analysis.

Technical Reference(s): OI-730, pg 6
SD-730- pg 37 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 30

Which one of the following is:

(1) The Minimum Technical Specifications required Fuel Pool water level? AND
(2) How is this level controlled?

A. (1) > 36 ft.
(2) A series of weirs controls the Fuel Pool minimum level and the maximum level is controlled by manually throttling makeup water.

B. (1) > 23 ft. above the top of the fuel racks.
(2) A series of weirs maintains a specific level and the maximum level is controlled by automatic level control of the Fuel Pool Skimmer Surge Tank.

C. (1) > 36 ft.
(2) A series of weirs maintains a specific level and the maximum level is controlled by automatic level control of the Fuel Pool Skimmer Surge Tank.

D. (1) > 23 ft. above the top of the fuel racks.
(2) A series of weirs controls the Fuel Pool minimum level and the maximum level is controlled by manually throttling makeup water.

Explanation (Optional):

A. Correct – The Tech Spec limit for FP level is >36 ft. A series of weirs controls the Fuel Pool minimum level the maximum level is controlled by manually throttling makeup water IAW OI-435, Sect 6.0.

B. Incorrect – This 23' above the top of fuel is the Technical Specifications for Reactor Pressure Vessel (RPV) Water Level during Refueling Operations above the fuel in the RPV. There is no automatic level control of the Fuel Pool Skimmer Surge Tank

C. Incorrect – There is no automatic level control of the Fuel Pool Skimmer Surge Tank

D. Incorrect – This 23' above the top of fuel is the Technical Specifications for Reactor Pressure Vessel (RPV) Water Level during Refueling Operations above the fuel in the RPV.

Technical Reference(s): 1C04B, A-4
OI-435, Sect 6.0. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 31

Which one of the following describes the design basis function of the Rod Worth Minimizer?

It enforces…

A. rod withdrawal with a programmed control rod sequence to limit the power excursion to prevent rapid dispersal of the fuel in the event of a Control Rod Drop Accident (CRDA)

B. control rod sequences designed to prevent exceeding the Minimum Critical Power Ratio when Reactor power is below 21.7% Rated Thermal Power

C. programmed rod movement that minimizes individual control rod worth to prevent exceeding the Maximum Extended Load Limit Analysis (MELLA) while in MODE 2

D. control rod sequences to limit the rate of heat production to < 280 calories/gram of fuel during control rod withdrawal when reactor power is > 21.7%.

Explanation (Optional):

A. Correct - Since the worth of an individual rod is highly dependent on core power distribution, rod sequence control provides a means of restricting the maximum reactivity insertion that could occur in a CRDA. The principal function of the NUMAC RWM is to limit rod motion such that high worth rods are not created, thereby limiting the maximum reactivity which could be added due to a control rod drop accident.

B. Incorrect – This is not a design function, the RWM does ensure that fuel operating limits are not exceeded and that the possibility of a high notch worth scram occurring is minimized.

C. Incorrect - This is not a design function, the RWM does ensure that fuel operating limits are not exceeded and that the possibility of a high notch worth scram occurring is minimized.

D. Incorrect - The RWM limits the rate of heat production to < 280 calories/gram of fuel during rod DROP accident NOT a control rod withdrawal. And the power level is when reactor power is <10%.

Technical Reference(s): SD-878.8, pg 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 32

The plant is operating in MODE 1 at 100% power with the following conditions:

- Turbine Building NSPEO reports that a very large lube oil leak has developed near the Main Generator
- Subsequent to the report alarm 1C07A A-7, TURBINE LUBE OIL BEARING HEADER LO PRESSURE activates
- The Turbine Building NSPEO reports that he cannot maintain Lube Oil Tank level

Which actions are required by AOP 693, Main Turbine/EHC Failures?

The ___(1)___ and the MSIV's shall be ___(2)___.

A. (1) Reactor will be scrammed then Main Turbine manually tripped
(2) closed

B. (1) Main Turbine will be tripped, and automatic Reactor scram verified
(2) closed

C. (1) Reactor will be scrammed then Main Turbine manually tripped
(2) left open

D. (1) Main Turbine will be tripped, and automatic Reactor scram verified
(2) left open

Explanation (Optional):

A. Correct – the reactor is scrammed, then the turbine is tripped, MSIV's are closed to facilitate breaking Main Condenser vacuum

B. Incorrect – the turbine is tripped before the reactor is scrammed, MSIV's are closed to facilitate breaking Main Condenser vacuum

C. Incorrect - the reactor is scrammed, then the turbine is tripped, MSIV's are closed to facilitate breaking Main Condenser vacuum

D. Incorrect - the turbine is tripped before the reactor is scrammed, MSIV's are closed to facilitate breaking Main Condenser vacuum

Technical Reference(s): AOP-693 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # # 20729

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 33

The plant is conducting a startup with the following conditions:

- The reactor is critical
- Reactor power is approximately 1%, 50 on range 8 of IRMs
- Reactor pressure is 950 psig
- The “A” Recirculation Pump has just tripped

With these plant conditions;
(1) Which one of the following indications must the Reactor Operator monitor?
(2) What is indicated by these indications?

A. (1) Excessive noise on the jet pump dP indicators
(2) Jet pump cavitations

B. (1) High flow indication on the operating loops jet pumps
(2) Jet pump cavitations

C. (1) Excessive noise on the jet pump dP indicators
(2) Cavitation of the operating recirculation pump

D. (1) High flow indication on the operating loops jet pumps
(2) Cavitation of the operating recirculation pump

Explanation (Optional):

A. Correct – IAW with OI-264, P & L 5and 10, at rated temperature and low reactor power (less than 2%), avoid single loop operation, even at minimum speed. If single loop operation is necessary for short periods of time, monitor jet pump flow to ensure cavitation does not occur. Jet pump cavitation is indicated by excessive noise on the jet pump dP indicators. In this question the plant is below 2% power (Range 8 0 on the IRMs and at rated pressure.

B. Incorrect - Jet pump cavitation is indicated by excessive noise on the jet pump dP indicators.

C. Incorrect – Recirc Pump cavitation is indicated by excessive vibration and sudden drop in pump discharge pressure and flow

D. Incorrect – Recirc Pump cavitation is indicated by excessive vibration and sudden drop in pump discharge pressure and flow

Technical Reference(s): OI-264, P & L 5and 10, pgs 4 & 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

5-09-11, Revised question

Proposed Question: RO Question # 34

The plant is operating in MODE 1 at 100% power with the following conditions:

- The FUEL POOL EXHAUST RADIATION MONITOR RIS-4131A Mode Switch is taken out of the OPERATE position by an I&C Technician

(1) Which one of the following initiations will occur?
(2) What action is required?

A. (1) Only the "A" Standby Gas Treatment system will initiate
(2) Verify the automatic isolation of the Secondary Containment ONLY

B. (1) Only the "A" Standby Gas Treatment system will initiate.
(2) Verify the automatic isolation of the Primary and Secondary Containment.

C. (1) Both Standby Gas Treatment systems will initiate.
(2) Verify the proper operation of SBGT, then operator may shutdown one train of SBGT

D. (1) Both Standby Gas Treatment systems will initiate.
(2) Verify the proper operation of SBGT, then operator must shutdown one train of SBGT

Explanation (Optional):

A. Incorrect - Both SBGT trains will automatically start. Secondary Containment will automatically initiate.

B. Correct - Both SBGT trains will automatically start.

C. Correct - The Pool exhaust high radiation of 8 mr/hr or mode switch out of operate will initiate both SBGT trains. If proper operation of SBGT is verified then proceed to Section 4.2 in order to place an activated SBGT Train in the Standby Mode, if desired.

D. Incorrect - It is NOT required to shutdown one train of SBGT.

Technical Reference(s): OI-170, pgs 8 and 9; SD-170; SD 959.1, page 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 35

The plant is operating in MODE 1 at 100% power with the following conditions:

- “A” and “B” Cooling Towers are in service
- A small nitrogen leak inside the shroud of the “E” Cooling Tower cell causes the deluge for the “E” and “F” Cells to initiate

Which one of the following describes the effect of this initiation on Cooling Tower operation?

The cooling tower fans will automatically ...

A. trip if running in “FWD”, but remain running if running in “REVERSE”

B. remain running unless high temperatures are confirmed by local temperature switches

C. trip if running in “FWD” or “REVERSE”. Taking the handswitch on 1C06 to “STOP” will reset the logic and allow the fan to be reset with no other operator actions

D. trip if running in “FWD” or “REVERSE”. The cooling tower deluge must be isolated and then reset in order to restart the fans

Explanation (Optional):

A. Incorrect - Activation of the Cooling Tower Deluge System automatically shuts off the associated tower fans.

B. Incorrect - Activation of the Cooling Tower Deluge System automatically shuts off the associated tower fans.

C. Incorrect - Activation of the Cooling Tower Deluge System automatically shuts off the associated tower fans when a pressure switch reads 6 psig pressure in the deluge system. The fan will not start until the pressure switch resets, meaning no pressure. The procedure isolates the deluge, then drains the deluge piping.

D. Correct - Activation of the Cooling Tower Deluge System automatically shuts off the associated tower fans when a pressure switch reads 6 psig pressure in the deluge system. The fan will not start until the pressure switch resets, meaning no pressure. The procedure isolates the deluge, then drains the deluge piping.

Technical Reference(s): OI-513, pg 4
ARP 1C06A A-5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 36

During a plant STARTUP with the reactor in MODE 2 the following conditions exist:

- The "B" IRM is selected to Range 3
- The "B" IRM/APRM Recorder switch is in the IRM position
- The "B" IRM fails upscale
- Annunciator 1C05B, B-3, IRM B, D, OR F UPSCALE TRIP OR INOP, alarms
- A “B” RPS half scram is received

The CRS directs the “B” IRM be bypassed. Which one of the following indications remain available?

1 - “B” IRM 1C05 indicating lamps on the Reactor Control Benchboard (EXCEPT bypass light)
2 - IRM "B" inputs to the IRM recorder
3 - "B" IRM outputs to the annunciators
4 - "B" IRM channel inputs to SPDS
5 - 1C36 alarm lights for the "B" IRM
6 - 1C36 meter indications for the "B" IRM

A. 1, 3, 4, 5

B. 2, 4, 5, 6

C. 1, 2, 4, 5

D. 2, 3, 5, 6

Explanation (Optional):

A. Incorrect - The IRM outputs to the indicating lamps on the Reactor Control Benchboard and IRM outputs to the annunciator are defeated.

B. Correct - When an IRM channel is bypassed, the following IRM functions are defeated:
a. The IRM UPSCALE trip to Reactor Protection System.
b. The IRM associated trips to the rod withdrawal block circuits of the Reactor Manual Control System.
c. The IRM outputs to the annunciator and sequence recorder.
d. The IRM outputs to the indicating lamps on the Reactor Control Benchboard. The Retract Permit Lamp will remain ON as long as the IRM channel is bypassed and the IRM detector is not full out.

C. Incorrect - The IRM outputs to the indicating lamps on the Reactor Control Benchboard are defeated.

D. Incorrect - The IRM outputs to the annunciator are defeated.

Technical Reference(s): OI-878.2, NOTE pg 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # # 20455

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 37

The plant is operating in MODE 1 at 100% power with the following conditions:

- Annunciator A-2 REACTOR BLDG SOUTH EAST AREA FLOOR DRAIN LEVEL HIGH alarms at panel 1C147, RB Floor Drain System Control
- Annunciator B-4 AREA WATER LEVELS ABOVE MAX NORMAL alarms at panel 1C14A, EOP Annunciators
- An operator reports from 1C21 that SE Corner Room level is slightly greater than 2 inches and rising very slowly.
- SANSOE reports from the SECR mezzanine that there is water on the floor and he will try to locate the leak

Which one of the following procedures:
(1) Shall be reported to the CRS as a possible entry, and
(2) What are the required actions

A. (1) AOP 902, Flood
(2) Scram the reactor and control level, pressure, reactor power.

B. (1) EOP 3, SECONDARY CONTROL
(2) Contact the Plant Chemist and have him sample the water prior to draining it to the Reactor Building Floor Drain Sump.

C. (1) AOP 902, Flood
(2) Contact the Radwaste Operator and have him pump down the Reactor Building Floor Drain Sump.

D. (1) EOP 3, SECONDARY CONTROL
(2) Have the Radwaste Operator open the affected valve to drain the area, and operate sump pumps as necessary.

Explanation (Optional):

A. Incorrect – AOP 902, Flood is for a Cedar River flood condition, not an internal water event. This AOP was chosen instead of EOP 1, since no RPV Control issues are part of this question. SE Corner Room level is above 2" but rising very slowly this is a case where there is a long time between Max Normal and Max Safe (10") therefore there is no entry condition for EOP 1.

B. Incorrect - There is no requirement to sample the water and time should not be spent in the EOP sampling the discharge of water from this area is required.

C. Incorrect – AOP 902, Flood is for a Cedar River flood condition, not an internal water event. This AOP was chosen instead of EOP 1, since no RPV Control issues are part of this question. SE Corner Room level is above 2" but rising very slowly this is a case where there is a long time between Max Normal and Max Safe (10") therefore there is no entry condition for EOP 1.

D. Correct - SE Corner Room level is slightly greater than 2 inches is above the Max Normal Operating Limit for the SE corner Room which requires an entry into EOP-3. The EOP requires operating available sump pumps to restore and maintain water level below the Max Normal Operating Limit

Technical Reference(s): EOP-3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 38

Which one of the following describes the relationship between INDICATED RPV water level on the Fuel Zone and GEMAC level instruments, and ACTUAL RPV water level when post accident conditions place Drywell Temperature and Reactor Pressure in the "Action is required" area of EOP-1, Graph 1, "RPV Saturation Temperature"?

All of the Fuel Zone level instruments read ___(1)___ and the GEMAC level instruments read ___(2)___.

A. (1) lower than actual
(2) lower than actual

B. (1) higher than actual
(2) higher than actual

C. (1) higher than actual
(2) lower than actual

D. (1) lower than actual
(2) higher than actual

Explanation (Optional):

A. Incorrect - a lack of understanding of reference leg and variable leg sensing lines affect from elevated drywell temperature, and the special compensation measures installed to counteract transient affects could lead to this conclusion

B. Correct - With Drywell parameters in the ACTION is Required” area of the curve a -23 inch penalty is applied to the Fuel Zone and the GEMAC This is done because the indicated level is higher than actual (See EOP 2 Caution 1)

C. Incorrect – The fuel zone and GEMACS all read higher

D. Incorrect - The fuel zone and GEMACS all read higher

Technical Reference(s): DAEC EOP 2 Bases Document, EOP Curves and Limits, pgs. 81-83, SD-880, pgs. 30-32,44-45 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: RO 95.00.00.14 (As available)

Question Source: Bank # WTS 10349

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Proposed Question: RO Question # 39

With the plant in MODE 5, REFUELING, and Core Alterations in progress.

- RPV level begins to lower unexpectedly

In accordance with Technical Specifications which of the following is the MINIMUM acceptable water level above the Reactor Vessel Flange, and the reason for that limit?

A. 23 feet to retain iodine fission product activity in the water in the event of a fuel handling accident and limit offsite doses from the accident to less than NRC Regulatory Guide limits.

B. 36 feet to ensure the time to boil assumptions for a loss of shutdown cooling are accurate.

C. 20' 1″ to retain iodine fission product activity in the water in the event of a fuel handling accident and limit offsite doses from the accident to less than NRC Regulatory Guide limits.

D. 36 feet to provide adequate shielding of drywell and refuel floor personnel during core alterations and limit dose exposure in the event of a fuel handling accident to less than NRC Regulatory Guide limits.

Explanation (Optional):

A. Correct – IAW TS LCO & Bases 3.9.6 - RPV water level shall be ≥ 23 ft above the top of the irradiated fuel assemblies seated within the RPV. The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel cavity. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than Regulatory Guide 1.183 limits.

B. Incorrect –This is not the reason for maintaining that level per TS.

C. Incorrect – This limit is related to the applicability of TS 3.9.7 for maintaining an operable RHR loop in SDC while in Mode 5.

D. Incorrect - This limit is related to the applicability of TS 3.9.7 for maintaining an operable RHR loop in SDC while in Mode 5. Although personnel dose limits are maintained lower with adequate RPV level during core alterations. This is not the reason for maintaining that level per TS.

Technical Reference(s): TS 3.9.6 Bases & LCO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Fermi

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 40

The plant was operating in MODE 1 at 98% power due to coastdown with the following conditions:

- A Loss of Vacuum event occurred
- A manual Reactor Scram was inserted
- All Control Rods are Full In
- Bypass Valves have failed closed.
- RPV Water Level is being maintained by Feedwater.
- Low Low Set is NOT working

Under these conditions stabilizing reactor pressure less than 1055 psig will ___(1)___ and minimize the effects of RPV level ___(2)___ on SRV openings.

A. (1) avoid repeated operation of the SRVs on high reactor pressure
(2) SHRINK

B. (1) allow the operator to manually reset the ATWS ARI/RPT logic if it initiated on high reactor pressure
(2) SHRINK

C. (1) avoid repeated operation of the SRVs on high reactor pressure
(2) SWELL

D. (1) allow the operator to manually reset the ATWS ARI/RPT logic if it initiated on high reactor pressure
(2) SWELL

Explanation (Optional):

A. Incorrect – This will avoid repeated operation of the SRVs on high reactor pressure however the concern with SRV openings is RPV level swell.

B. Incorrect – manual reset of the scram would be possible NOT ATWS ARI/RPT logic. The concern with SRV openings is RPV level swell.

C. Correct – Per EOP 1 Bases - Swell resulting from SRV actuation may result in high level trips of steam driven systems even if level is maintained low in the normal band. It may then be necessary to define a wider control band to maintain level below the high level trip setpoint. Bases for RC/P-4 step " Stabilize RPV pressure Below 1055 psig" - The direction to stabilize RPV pressure in Step RC/P-4 means to limit changes in RPV pressure (both increases and decreases) to within as small a band as possible. Controlling RPV pressure below this value avoids SRVs lifting on high pressure and allows the scram logic to be reset (provided no other scram signal exists).

D. Incorrect - manual reset of the scram would be possible NOT ATWS ARI/RPT logic.

Technical Reference(s): EOP 1 bases page 24 and 55 (Rev 14) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 41

The plant was operating at rated power when a DBA LOCA occurred.

Under these conditions, ___(1)___ could cause the drywell to exceed its ___(2)___ design pressure limit.

A. (1) a Torus to Drywell Vacuum Breaker failing OPEN
(2) internal.

B. (1) a Torus to Drywell Vacuum Breaker failing CLOSED
(2) external

C. (1) a Reactor Building to Torus Vacuum Breaker failing OPEN
(2) external

D. (1) a Reactor Building to Torus Vacuum Breaker failing CLOSED
(2) internal

Explanation (Optional):

A. Correct – IAW SD 959 - Containment Characteristics after LOCA with Torus /Drywell Vacuum Breaker Failed Open - Steam flows from the drywell to the torus through the vacuum breaker equalizing the pressure. The steam is not forced through the downcomers and up through the water, but instead is dumped on the surface of the water in the torus. As a result, the drywell pressure will probably exceed design pressure.

B. Incorrect - In this condition, drywell pressure could lower and cause the Torus to Drywell differential pressure to exceed 2 psid.

C. Incorrect - correct if the vacuum breaker failed closed

D. Incorrect – IAW SD 959 page 25, if a reactor building to torus vacuum breaker were to be failed closed in the case of a DBA, there would be little effect. The purpose of the reactor building to torus vacuum breakers is to ensure that neither the torus nor drywell exceed their external pressure limit.

Technical Reference(s): SD 959 rev 4 page 24 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 42

The plant is operating in MODE 1 at 100% power with the following conditions:

- A Main Turbine trip occurs

How is the extraction steam system affected?

The High Pressure Extraction Drain to Condenser, CV-1237, ___(1)___ and 2nd stage reheat steam high and low load valves ___(2)___.

A. (1) opens
(2) open

B. (1) closes
(2) close

C. (1) opens
(2) close

D. (1) closes
(2) open

Explanation (Optional):

A. Incorrect - 2nd stage reheat steam high and low load valves close

B. Incorrect - The High Pressure Extraction Drain to Condenser, CV-1237, opens

C. Correct – IAW SD 646 page On any Main Turbine trip, High Pressure Extraction Drain to Condenser CV-1237 opens and 2nd stage reheat steam high and low load valves close. These actions result from the trip of the Turbine Extraction Relay Dump Valve, which isolates and vents off control air that is required for these valves to be open.

D. Incorrect - The High Pressure Extraction Drain to Condenser, CV-1237, opens and the High Pressure Extraction Drain to Condenser, CV-1237, close

Technical Reference(s): SD 646 Rev.10 page 33 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 43

The plant is operating in MODE 1 at 35% power with the following conditions:

- The "A" Circ Water Pump is in operation
- The “A” Cooling Tower is in operation

Assuming no operator action, which of the following conditions would cause the “A” Circ Water Pump to trip?

A: Circ Water Pit level lowering to 13 ft

B: Losing 1Y11, Instrument AC Division 1

C: Losing 1Y23, 120 VAC Uninterruptible power supply

D: Closing MO-4208, HP CONDENSER 1E-7B SOUTH WATER BOX OUTLET

Explanation (Optional):

A: There is an administrative limit of 48 hours of operation with Circ Pit level below 13 ft

B: Loss of 1Y11 power will cause KY-4201 to be deenergized allowing stored energy in the accumulators to be released and close HO-4201 which trips 1P-4A. None of the malfunctions listed is a direct trip of a Circ Pump. All require knowledge of system interactions

C: There is a circ pump trip caused by a loss of 1Y11/1Y21 which can be confused with 1Y23

D: MO4208 is a starting interlock, not a trip

Technical Reference(s): OI-442 "Circulating Water System" Rev. 81, P&L #7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 32.02.02.02 (As available)

Question Source: Bank # WTS 10375

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 44

The plant is operating in MODE 1 at 100% power with the following conditions:

- The “B” Reactor Recirculation Pump tripped
- All systems responded as designed

Which of the following describes the INITIAL reactor water level response and why?

Indicated reactor water level will ___(1)___ due to the ___(2)___.

A. (1) RISE
(2) collapse of steam voids

B. (1) LOWER
(2) lack of coolant velocity to sweep voids into the steam separator

C. (1) RISE
(2) displacement of water by increased steam voiding

D. (1) LOWER
(2) initial delay in feedwater control system response

Explanation (Optional):

A. Incorrect – steam voiding would increase

B. Incorrect – steam voiding would increase

C. Correct - the trip of the pump would result in more steam voiding. RPV would increase until the FW control system restored level to the normal value

D. Incorrect – level would increase due to increased voiding

Technical Reference(s):	GFES Chapter 8, Operational Physics, discussion on RR flow and Reactor Power (discussion is to increase RR flow, this question is reversed)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	WTS 1109	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 45

The plant is operating in MODE 1 at 100% power with the following conditions:

- “A” RHR loop is tagged out of service for maintenance
- A fire has been verified in the turbine building, in Fire Area TB1

Which of the following is an action that is required IAW AOP 913, Fire, and why?

Dispatch an NSPEO to ____.

A. manually close MO-1905, RHR LOOP B LPCI INBD INJECT ISOL if it spuriously opens to prevent RPV injection when not required.

B. manually open MO-1905, RHR LOOP B LPCI INBD INJECT ISOL if only “B” RHR is available to ensure an RPV injection path.

C. manually open V-19-48, RHR LOOP CROSSTIE to ensure an RPV injection supply if only “B” RHR is available for RPV injection.

D. manually open BOTH V-19-48, RHR LOOP CROSSTIE and MO-1905, RHR LOOP B LPCI INBD INJECT ISOL to ensure an RPV injection supply if only “B” RHR is available for RPV injection.

Explanation (Optional):

A. Incorrect – no actions listed in AOP 913 to manually close the valve.

B. Correct – IAW AOP 913 Path TB1 continuous recheck statement

C. Incorrect - the direction is to CLOSE the V-19-48 valve (RB3 Continuous Recheck Statement, page 83)

D. Incorrect – the direction is to CLOSE the V-19-48 valve (RB3 Continuous Recheck Statement, page 83)

Technical Reference(s): AOP 913 Path TB1 continuous recheck statement (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 46

Which of following describes why achieving COLD SHUTDOWN BORON WEIGHT is desired during EOP-ATWS mitigation actions?

A. To assure that the reactor will remain shutdown prior to raising RPV level to 170” to 211”.

B. To assure that the reactor will remain shutdown irrespective of control rod position and with RPV water level at a minimum of -25″.

C. To assure that the reactor will remain shutdown under all conditions so a reactor cooldown can begin.

D. To assure that the reactor will remain shutdown with RPV water level at a minimum of -25”.

Explanation (Optional):

A. Incorrect – this is the concept of Hot Shutdown Boron Weight

B. Incorrect - this partially defines Hot Shutdown Boron Weight. RPV level must be in the normal band

C. Correct – IAW EOP ATWS Bases, page 68 – "Injection of the Cold Shutdown Boron Weight (CSBW) of boron into the RPV ensures that the reactor is shutdown and will remain shutdown. The CSBW is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions."

D. Incorrect - this partially defines Cold Shutdown Boron Weight but with the incorrect RPV level.

Technical Reference(s): EOP ATWS Bases Rev.14 page 68 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 47

The plant is operating in MODE 1 at 100% power with the following conditions:

- 1K1 is in STANDBY mode
- A loss of Instrument Air header pressure occurs
- Instrument Air header pressure is 90 psig and lowering slowly

Which one of the following is:
(1) The reason the Backup Air Compressor 1K1 starts at this time?
(2) What system will supply Backup Air Compressor 1K1 cooling?

A. (1) To supply ONLY the Instrument Air Header pressure.
(2) Compressor Cooling Water System

B. (1) To supply BOTH the Instrument & Service Air Headers
(2) Compressor Cooling Water System

C. (1) To supply ONLY the Instrument Air Header pressure.
(2) Well Water System

D. (1) To supply BOTH the Instrument & Service Air Headers
(2) Well Water System

Explanation (Optional):

A. Incorrect – initially both service and instrument air headers are supplied.

B. Incorrect – The 1K1 is supplied by Well water.

C. Incorrect - initially both service and instrument air headers are supplied.

D. Correct – Unless header pressure drops to 82 psig, both headers are supplied. The well water system is the primary cooling water medium for the 1K1

Technical Reference(s): AOP 518
SD 518 Rev 8. pages 13,14,24,27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 48

The plant is operating at rated power. The “A” SBDG is in service for a scheduled surveillance test.

Then, a loss of all River Water Supply (RWS) Pumps occurs.

The plant is manually scrammed and the initial actions of IPOI-5 are completed successfully.

Which of following describes RWS system loads that are DIRECTLY impacted and an action required IAW AOP 410, Loss of River Water Supply.

Monitor ___(1)___ system loads and ___(2)___.

A. (1) ESW, RHRSW and GSW
(2) Secure the running SBDG

B. (1) Circ Water, RHRSW, and Fuel Pool Cooling
(2) Secure the running SBDG

C. (1) ESW, RHRSW and GSW
(2) Open the Circ Water Inlet to Blowdown Line valve MO-4253 to maintain Circ Water Pit inventory.

D. (1) Circ Water, RHRSW, and Fuel Pool Cooling
(2) Open the Circ Water Inlet to Blowdown Line valve MO-4253 to maintain Circ Water Pit inventory.

Explanation (Optional):

A. Correct – IAW AOP 410 page 4 step 7 - Shutdown any SBDG not required to ensure one Essential Bus is energized and/or required to ensure adequate core cooling. IAW SD 410 – RWS Purpose - to provide makeup water from the Cedar River for the Circulating Water System, GSW, RHRSW, ESW, Fire System and Radwaste Dilution Systems to replace that which is lost due to evaporation, blowdown and normal uses.

B. Incorrect – Fuel Pool Cooling is not directly impacted by this loss. It is cooled by RBCCW

C. Incorrect - The Circ Water Inlet to Blowdown Line valve MO-4253 is required to be CLOSED.

D. Incorrect - Fuel Pool Cooling is not directly impacted by this loss. It is cooled by RBCCW. The Circ Water Inlet to Blowdown Line valve MO-4253 is required to be CLOSED.

Technical Reference(s): AOP 410 Rev.14 page 4
SD 410 – system purpose (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 49

The plant is operating in MODE 1 at 100% power with the following conditions:

- ITC Midwest notifies the Main Control Room of a degraded offsite power condition
- 1A3 and 1A4 bus voltage is continuing to degrade toward a trip condition
- 1A3 and 1A4 have not yet tripped

Which of the following is required IAW AOP 304 – Grid Instability?

A.
(1) Start the SBDGs
(2) Parallel and load the Essential Buses
(3) Reduce Recirc to 27 mlbm/hr Flow
(4) Scram the reactor

B.
(1) Reduce Recirc to 27 mlbm/hr Flow
(2) Scram the reactor
(3) Start the SBDGs
(4) Parallel and load the Essential Buses before the 1A3 and 1A4 bus supply breakers trip

C.
(1) Reduce Recirc to 27 mlbm/hr Flow
(2) Scram the reactor
(3) Do not attempt to start and load the SBDGs
(4) Continue to monitor for Grid Instabilities

D.
(1) Start the SBDGs
(2) Do NOT parallel and load the Essential Buses
(3) Continue to monitor for Grid Instabilities
(4) If the 1A3 and 1A4 trip, verify the SBDGs load their respective buses and the Reactor Scrams

Explanation (Optional):

A. Incorrect – would not start the SBDGs with degraded conditions

B. Incorrect – would continue to monitor grid instability and continue with IPOI 5 actions. would not start the SBDGs.

C. Correct – IAW AOP 304 Caution - It is not appropriate to manually start and load a SBDG during degraded grid conditions. Followup action 1.b. - **IF** It appears that busses 1A3 and 1A4 will trip due to degrading grid conditions. Reduce Recirc to 27 mlbm/hr and Flow Scram the reactor.

D. Incorrect - would not start the SBDGs with degraded conditions

Technical Reference(s): AOP 304 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 50

IPOI-5, Reactor Scram, has been entered and plant conditions are as follows:

- Level setback pushbutton has been depressed
- Scram choreography is complete
- The Feedwater Master Controller, LC-4577, is in AUTO
- RPV level has risen to 175 inches and is stable

The CRS directs that RPV level be returned to the green band (186″ to 195″).

Which one of the following describe the MINIMUM actions required to return reactor water level to the normal band IAW IPOI-5, Reactor Scram?

A. Adjust the Feedwater Master Controller LC-4577 in AUTO until reactor level is restored to the normal band.

B. Place the Feedwater Master Controller, LC-4577, to MANUAL and adjust flow to return level to the normal band. LC-4577 should remain in MANUAL.

C. Reset the Setpoint Setback by depressing the reset pushbutton on 1C05 and then adjusting the Feedwater Master Controller LC-4577 AUTO setpoint until level is in the normal band.

D. Place the "A" and "B" Feedwater Regulating Valve Controllers in MANUAL and adjust flow until level is restored to the normal band. Then place those controllers back in AUTO.

Explanation (Optional):

A. Correct – IAW IPOI 5 - Use any or all of the following techniques as necessary to control RPV level: After RPV level starts to rise as indicated on the wide range

Yarways, then place Master Feed Reg controller LC-4577 in MANUAL and close the Feed Reg valves. Restore LC-4577 back to AUTO after RPV level stabilizes.

B. Incorrect – The minimum actions would be to leave the controller in AUTO, and the procedure requires the controller be set back to AUTO.

C. Incorrect – The Feedwater Master Controller, LC-4577, must be in manual to take the setback circuit out of the level control system .

D. Incorrect – not required to place the FRV controllers in manual

Technical Reference(s): IPOI 5 Rev 54 step 3.2 (4) a. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: RO-45.05.01.05-05 (As available)

Question Source: Bank # 20086

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 51

In a LOCA event, which of the following is a concern if a Torus Water lowered to a level of 5.8 feet?

(1) What is the specific equipment issue at this elevation?
(2) What are the implications of this equipment being uncovered?

A. (1) The HPCI Turbine Exhaust will become uncovered
(2) This will directly pressurize the torus. The consequences of not doing so may result in failure of the primary containment from over pressurization

B. (1) The HPCI Turbine Exhaust will become uncovered
(2) To ensure that steam discharged from the drywell into the torus following a primary system break will be adequately condensed. If a primary system break were to occur with torus water level below the bottom of the HPCI Turbine Exhaust, pressure suppression capability would be unavailable and torus pressure could exceed the Primary Containment Pressure Limit.

C. (1) The downcomer vent openings will become uncovered
(2) This will directly pressurize the torus. The consequences of not doing so may result in failure of the primary containment from over pressurization

D. (1) The downcomer vent openings will become uncovered
(2) To ensure that steam discharged from the drywell into the torus following a primary system break will be adequately condensed. If a primary system break were to occur with torus water level below the bottom of the downcomers, pressure suppression capability would be unavailable and torus pressure could exceed the Primary Containment Pressure Limit.

Explanation (Optional):

A. Correct – EOP 2 Bases Step TL/6 – (1) A torus level of 5.8 feet corresponds to the HPCI turbine exhaust elevation.
(2) Operation of the HPCI system with its exhaust device not submerged will directly pressurize the torus. HPCI operation is therefore secured when torus level cannot be maintained above 5.8 feet to preclude pressurizing the torus. The consequences of not doing so may result in failure of the primary containment from over pressurization. Thus, HPCI must be secured irrespective of adequate core cooling concerns.

B. Incorrect – (1) The HPCI turbine exhaust level is 5.8 feet (correct), however (2) the discussion is the bases discussion for the 7.1 ft torus level.

C. Incorrect - (1) The downcomers vent openings are at 7.1 ft torus level. (2) The discussion is the bases discussion for the 5.8 ft torus level.

D. Incorrect – (1) The downcomers vent openings are at 7.1 ft torus level. (2) The discussion is the bases discussion for the 7.1 ft torus level.

Technical Reference(s): EOP 2 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 52

The plant is in Mode 4 with RHR "A" in shutdown cooling with the following conditions:

- RPV water level momentarily drops to 168 inches and is recovered to 173 inches

What is the effect on Shutdown Cooling?

A. Shutdown Cooling remains in service.

B. The “A” RHR pump trips directly due to RPV level. The inboard and outboard Shutdown Cooling Isolation valves go CLOSED.

C. The “A” RHR pump remains in service but only on minimum flow. The inboard and outboard Shutdown Cooling Isolation valves go CLOSED.

D. The “A” RHR pump tripped because a loss of suction path was sensed by the pump trip circuitry when the Shutdown Cooling Isolation valves began to CLOSE.

Explanation (Optional):

A. Incorrect – The pump tripped and the valves closed

B. Incorrect – The pump tripped due to loss of suction path NOT low RPV level

C. Incorrect - The pump tripped and the valves closed

D. Correct – The valves close at 170” RPV level. When they begin to close (not fully open) the pump trips because a loss of suction path is sensed by the pump trip circuitry.

Technical Reference(s): SD 149 Rev.11. pages 11, 32, Figure 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 10960

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 53

The plant is operating in MODE 1 at 100% power with the following conditions:

- Annunciator 1C03A A-4, OFFGAS VENT PIPE RM-4116A/B HI-HI RAD alarms
- Standby Gas Treatment System initiates

Which of the following choices below could be the source for the above alarm?

(1) A Reactor Recirc pump seal leak
(2) A Condenser Bay steam leak
(3) A RWCU Pump seal leak
(4) A leak in the Torus Room

A. (1), (2) and (3)

B. (2), (3) and (4)

C. (1), (3) and (4)

D. (1), (2) and (4)

Explanation (Optional):

A. Incorrect – (1) would be contained in the drywell

B. Correct – See SD 733 Figures 4,5,6

C. Incorrect - (1) would be contained in the drywell

D. Incorrect - (1) would be contained in the drywell

Technical Reference(s): SD 733 Figures 4,5,6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Proposed Question: RO Question # 54

During execution of ATWS-RPV Control, it is required to lower RPV Water Level to at least 87 inches.

Which of the following describes the reason for this requirement?

It is required to lower RPV Water Level to at least 87 inches to _______.

A. uncover the fuel to reduce natural circulation and limit the peak power level to below the fuel thermal limits

B. uncover the feedwater spargers to reduce subcooling and limit the onset of reactor power / core flow instabilities

C. isolate RWCU to prevent boron removal by the system and limit the peak power level to below the fuel thermal limits

D. trip the operating Recirculation Pumps to reduce forced circulation and limit the onset of reactor power / core flow instabilities

Explanation (Optional):

A. Incorrect - 87 inches will NOT uncover fuel.

B. Correct – IAW EOP ATWS Bases Continuous Recheck Statement - The conditions expressed in this Continuous Recheck Statement, combined with the inability to shutdown the reactor through control rod insertion, dictate a need to promptly reduce reactor power in order to prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities. This is accomplished by transferring to entry point 7 and lowering RPV water level to +87 inches in Step /L-2. An RPV water level of +87 inches is 2 feet below the lowest nozzle in the feedwater sparger. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling.

C. Incorrect - RWCU is verified isolated, but the reason for lowering level to 87 inches is NOT based on RWCU automatic isolation at 119.5 inches

D. Incorrect - RR Pumps will be verified tripped if power is above 5%, but the reason for lowering level to 87 inches is NOT based on RR Pump ATWS RPT at 119.5 inches.

Technical Reference(s): EOP ATWS Bases Rev 14 page 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 11294

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 55

A transient resulted in the following plant conditions:

- RPV level is 60 inches and steady
- RPV pressure is 800 psig and lowering slowly
- Torus and Containment Sprays have been initiated once
- Drywell Pressure is 1.6 psig and steady
- Drywell Temperature is 100°F and steady
- Torus Temperature is 102°F rising slowly

The Control Room Supervisor directs the operator to maximize torus cooling. Is this allowed by current plant conditions? Why or why not?

A. Yes, since adequate core cooling has been assured, the operator may establish Torus Cooling.

B. Yes, since there is less than a 2 psig drywell pressure signal, the operator may establish Torus Cooling.

C. No, since RPV level is less than 64" and drywell pressure is less than 2 psig, Torus Cooling may NOT be established.

D. No, since RPV pressure is 800 psig and LPCI loop select has selected a loop, Torus Cooling may NOT be established.

Explanation (Optional):

A. Incorrect – There is precaution on verifying adequate core cooling and with 60” in the RPV adequate core cooling is assured, however the torus cooling valves cannot be opened with less than 2 psig in the drywell and the LPCI signal still in.

B. Incorrect – The torus cooling valves cannot be opened with less than 2 psig in the drywell and the LPCI signal still in.

C. Correct – IA OI-149, Sect 5.3, pg 32, The Containment Spray and Cooling valves are interlocked closed when Drywell pressure is < 2 psig with a LPCI Initiation signal present. The LPCI signal is still present because the RPV water level is <119.5 inches.

D. Incorrect – Torus cooling could still be placed in service with these conditions IF DW pressure was >2psig.

Technical Reference(s): OI-149, pg 32 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 19019

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 56

The plant is operating in MODE 1 at 100% power when the following alarm occurs:

- 1C08A C-8, INSTRUMENT AC 1Y11 UNDERVOLTAGE OR INVERTER TROUBLE

What is the plant response to this annunciator?

If the alarm was caused by a ____.

A. low inverter AC OUTPUT, the RWCU system will isolate.

B. low inverter AC OUTPUT, the RWCU pumps will trip but the system will NOT isolate.

C. low voltage condition on Instrument Bus 1Y11, the “A” Recirc Pump will trip.

D. low voltage condition on Instrument Bus 1Y11, the “A” Recirc Pump scoop tube will lock up.

Explanation (Optional):

A. Incorrect – Low AC output results only in a trouble lamp on 1D15

B. Incorrect - Low AC output results only in a trouble lamp on 1D15

C. Incorrect – The pump does not trip but the scoop tube locks up

D. Correct – IAW ARP 1C08A C-8, Section 2.2, If the cause was due to a low voltage condition on the bus - RWCU Pumps 1P-205A and B trip, RWCU System isolates and Recirc Pump 1P-201A scoop tube locks up As Is.

Technical Reference(s): 1C08A C-8 Sections 1 and 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 57

The plant was operating in MODE 1 at 100% power when a NON-FIRE event occurred that required evacuation of the Control Room per AOP-915, Shutdown Outside the Control Room.

The following actions have been completed:

- Manual SCRAM has been inserted.
- ALL RODS have been verified inserted using the "One Rod Permissive" technique.
- The 1C05 operator has completed the "as time permits" actions of AOP-915 and evacuated the Control Room.

When the 1C05 Operator left the control room the Mode Switch would be in_____.

A. RUN

B. REFUEL

C. SHUTDOWN

D. START & HOT STBY

Explanation (Optional):

A. Correct –AOP 915 requires Reactor Mode Switch placed in RUN following Reactor Scram actions.

B. Incorrect - REFUEL position was used to verify ALL RODS IN.

C. Incorrect - SHUTDOWN is the normal post-scram Mode Switch position.

D. Incorrect - START & HOT STBY may be selected if the candidate knows a position other than SHUTDOWN is used, but doesn't know the correct position.

Technical Reference(s): AOP 915 Rev.41, Step 4.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 94.28.01.03 (As available)

Question Source: Bank # WTS

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 58

In accident conditions, IAW EOP-2, Primary Containment Control, action is required if drywell temperature cannot be restored and maintained below 280°F.

Why is action required at this step of the EOP-2?

A. At this temperature, closure of the MSIVs, if required, could not be assured because the MSIV Solenoids have reached their environmental qualification temperature limit.

B. Implementation of Drywell Spray above this temperature will NOT prevent exceeding the drywell analytical withstand temperature.

C. To provide margin to the temperature where the ADS SRVs and ADS Solenoids may not function if required to depressurize to RPV.

D. Torus to Drywell Vacuum Breakers are not designed to operate at this temperature and may not be able to function and minimize a Torus pressure spike under LOCA conditions.

Explanation (Optional):

A. Incorrect – The MSIVS and their solenoids are not a concern at this point in the EOPs. They are in all probability already closed due to a LOCA condition.

B. Incorrect – Drywell Spray if not already initiated may prevent exceeding the drywell analytical withstand temperature however the EOPs require an ED in this case for that purpose

C. Correct - IAW EOP-2 Bases - The EQ rating of equipment in the drywell, specifically the ADS valves and ADS solenoids, is 340 °F for a significant time. Although EQ analysis indicates that the ADS valves are operable for an extended period of time at 340 °F, management expectation is that operators will direct ED before 340 °F to ensure that the EQ limits and the drywell analytical withstand temperature is not exceeded.

D. Incorrect – the design temperature of the Drywell is 281F

Technical Reference(s): EOP-2 Bases Rev.13 page 41 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Proposed Question: RO Question # 59

The plant is in MODE 5 when a fuel handling accident occurs with the following conditions:

- No OPDRVs are in progress
- No PCIS Group III isolation setpoints have been exceeded during the event
- The “A” Standby Gas Treatment System is manually initiated with isolation IAW OI-170, Standby Gas Treatment System
- Secondary Containment Isolation Damper 1V-AD-19A fails to close

What is the operational implication of this condition?

Possible ____ .

A. entry into LCO 3.0.3 due to loss of Secondary Containment

B. possible release via Reactor Building Exhaust Fans 1VEF11A or 1VEF11B

C. excessive flow thru the operating SBGT train

D. unfiltered release from the Secondary Containment

Explanation (Optional):

A. Incorrect – loss of Secondary Containment is not an LCO 3.0.3 issue

B. Incorrect – A Group 3 isolation signal will trip the 11A & 11B fans

C. Incorrect – SBGT have flow controllers to control the flow going thru the SBGT train

D. Correct – With only one division of the Group 3 in, and one isolation damper failed to close, there is a possibility of unfiltered release from the Secondary Containment thru the open isolation damper.

Technical Reference(s): SD 733 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	WTS 11401	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 60

Which one of the following describes how primary containment Hydrogen and Oxygen concentrations are monitored?

	O_2 Concentration	H_2 Concentration	Can H_2/O_2 Analyzers be used during Normal Operations?	Can H_2/O_2 Analyzers be used during Emergency Operations?
A.	Not normally monitored	Not normally monitored	No	Yes
B.	Continuously monitored	Continuously monitored	Yes	Yes
C.	Continuously monitored	Not normally monitored	Yes	Yes
D.	Not normally monitored	Continuously monitored	Yes	No

Explanation (Optional):

A. Incorrect – H_2O_2 Analyzers may be used for both H_2 and O_2 monitoring during both normal operation and emergencies. O2 is normally monitored.

B. Incorrect - H2 does not have a stand alone detector. H_2O_2 Analyzers may be used for both H_2 and O_2 monitoring during both normal operation and emergencies.

C. Correct – IAW SD 573 - There is only one oxygen detector and it is associated with the "B" loop of CAMS. In the event the detector should become unavailable during normal operations, then the H_2-O_2 Analyzer(s) could be run to verify that containment oxygen is meeting the technical specification requirement.

D. Incorrect - H2 does not have a stand alone detector. H_2O_2 Analyzers may be used for both H_2 and O_2 monitoring during both normal operation and emergencies.

Technical Reference(s): SD 573 Rev.10 page 34 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 61

The plant is in MODE 3, Shutdown Cooling is in service with "A" RHR Pump in service

- Reactor coolant temperature and pressure are slowly rising.
- RPV level is 190 inches stable, maintaining on dump flow

The Shutdown Cooling automatic isolation actions have all occurred as designed.

The reason for these automatic actions is to prevent ____.

A. RHR suction piping overpressurization

B. steam voiding in the RHR pump seals

C. overpressurizing the RHR pump seals

D. establishing a drain path from the RPV to the torus

Explanation (Optional):

A. Correct – The Reactor Steam Dome Pressure — High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System (i.e., the shutdown cooling suction valves). This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario (i.e., a break of the low pressure RHR suction piping caused by exposure to relatively high pressure RPV fluid)

B. Incorrect – this would not be a primary concern

C. Incorrect – overpressurizing the piping is the concern

D. Incorrect – there are valve interlocks that prevent this from occurring.

Technical Reference(s): TA Bases 3.3.6.1 6.a. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 10569

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 62

A loss of coolant accident has occurred. The following plant conditions exist:

- Reactor Water Level +110 inches and slowly rising
- Drywell Pressure is 2.5 psig and slowly lowering
- Torus Temperature is 110 degrees F. and slowly rising
- The Essential Buses are being powered from the Standby Transformer
- A & B ESW pumps are in service
- A, B and C RHR pumps are in service
- A, B, and D RHRSW pumps are in service

Which one of the following describes the actions required, in order, to place the "D" RHR pump in Torus Cooling?

1. Place HS-1903C Enable Containment Spray Valves in the MAN position.
2. Close MO-1940, "B" Heat Exch Bypass Valve
3. Remove from service either the "B" RHR pump OR the "B" RHRSW pump OR the "D" RHRSW pump.
4. Throttle MO-1934, Torus Cooling Test Valve, to maintain 4800 gpm per each operating RHR pump.
5. Open MO-1932, Outboard Torus Cooling/Spray Valve.
6. Start the "D" RHR pump.

A.	B.	C.	D.
3.	6.	3.	6.
6.	5.	6.	5.
1.	1.	5.	2.
5.	3.	4.	4.
4.	2.	2.	1.
2.	4.	1.	

Explanation (Optional):

A. Correct – IAW OI 149 QRC 2 – **CAUTION** While the Essential buses are powered from the Standby Transformer, do not run more than a total combination of 3 RHR/RHRSW pumps on each essential bus. (e.g. 2 RHR pumps & 1 RHRSW pump , or 1 RHR pump & 2 RHRSW pumps). With a combination of 3 RHR/RHRSW pumps in service, stop one pump before starting the out of service pump.
If a LPCI HI Drywell pressure condition (2 #) exists**, place HS-2001C[1903C]** Enable Containment Spray Valves in the **MAN** position.

Actions are listed in the order of the QRC

B. Incorrect – one of the listed pumps must first be removed from service

C. Incorrect – the Enable Containment Spray Valves HS must be in the MAN position

D. Incorrect – one of the listed pumps must first be removed from service and the Enable Containment Spray Valves HS must be in the MAN position

Technical Reference(s): OI 149 QRC 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Proposed Question: RO Question # 63

The plant is operating in MODE 1 at 100% power with the following conditions:

- A large leak in the Drywell from the RBCCW System occurs
- A fast power reduction is performed IAW IPOI-4, Shutdown
- The reactor is manually scrammed
- Drywell and Reactor Building Sump High Sump Level alarms are IN
- All scram signals are clear and the scram is reset

Assuming no other operator actions have been taken, which of the following is correct concerning these conditions?

A. The Reactor Building Equipment Drain Sump is filling from the Scram Discharge Volume header and pumps will transfer water to Radwaste with no further operator action.

B. The Reactor Building Floor Drain Sump is filling from the Scram Discharge Volume header and pumping down to the Floor Drain Collector Tank.

C. The Drywell Equipment Drain Sump is filling from the RBCCW leak and pumps will transfer water to Radwaste with no further operator action.

D. The Drywell Floor Drain Sump is filling from the RBCCW leak and pumping down to the Floor Drain Collector Tank.

Explanation (Optional):

A. Correct – IAW SD 920-1, the CRD Hydraulic system drains to the reactor building equipment drain sump. When the scram is reset, the SDV will drain to that sump and pump to the radwaste collector tank.

B. Incorrect – The SDV does not drain into the floor drain

C. Incorrect – The Drywell Equipment drain would be isolated and not pumping down until PCIS Isolation signal was clear and reset.

D. Incorrect – The Drywell Floor drain would be isolated and not pumping down until the PCIS group 2 signal was clear and reset.

Technical Reference(s): SD 920-1 Rev.4, page 18, figures 1,2,5,6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Proposed Question: RO Question # 64

The plant is operating in MODE 1 at 100% power with the following conditions:

- The “A” CRD pump out of service to replace the motor bearings
- The 1A4 bus suffers a lockout trip and is de-energized

Due to a loss of drywell cooling, the CRS directs a manual reactor scram.

What will be the effect on the control rods?

A. Control rods will NOT insert on the scram. EOP-1 will be entered and transferred to EOP-ATWS for actions to be directed. Actions directed will be for a HIGH power ATWS.

B. Control rods will insert on the scram. EOP-1 will be entered and then IPOI-5.

C. Control rods will NOT fully insert on the scram. EOP-1 will be entered and transferred to EOP-ATWS for actions to be directed. Actions directed will be for a LOW power ATWS.

D. Control rods will insert on the scram. EOP-1 will be entered and then IPOI-5. A CRD pump must be re-started before the scram is able to be reset.

Explanation (Optional):

A. Incorrect – Control rods will insert into core without CRD pump running. This answer is plausible if the candidate believes that the rods will be stuck full out.

B. Correct – Control rods insert without CRD pump, EOP 1 will be required to be entered on the RPV level shrink, and IPOI-5 is the scram procedure.

C. Incorrect – Control rods will insert into core without CRD pump running. This answer is plausible if the candidate believes that the rods will partially insert, but not go full in.

D. Incorrect – CRD pump is not required to reset the scram.

Technical Reference(s): IPOI-5 (reset scram section) SD 255 (ball check valve discussion) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC 19984

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and functions of reactivity control mechanisms and instrumentation.

Comments:

Proposed Question: RO Question # 65

When carrying out RPV FLOODING EOP with 62 control rods not full in, what is the required position of the Main Steam Isolation Valves (MSIVs), and what is the reason for that requirement?

Main Steam Isolation Valves are required to be ____.

A. open, to allow Main Steam flow to assist in rapidly depressurizing the RPV and ensure boron is mixed throughout the vessel.

B. open, to allow flooded RPV indications to be obtained from Main Steam Line Flow Instruments.

C. shut, the ONLY concern is to avoid excessive water inventory loss from the RPV during flooding.

D. shut, to ensure adequate boron concentration in the vessel and avoid damage to downstream equipment.

Explanation (Optional):

A. Incorrect – The MSIVs are shut in EOP-ATWS step RPV/F-12. Boron would be diluted if the MSIVs were open

B. Incorrect – The MSIVs are shut per the EOP

C. Incorrect – Inventory loss is not the concern.

D. Correct – The MSIVs are shut in EOP-ATWS step RPV/F-12. IAW the bases, If the MSIVs were not closed, boron would be lost from the RPV when water level reached the elevation of the main steam lines. Leaving the MSIVs open would also risk damage to downstream equipment that might be needed during later recovery actions.

Technical Reference(s): EOP-ATWS
EOP-ATWS Bases Rev 12 page 23 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Proposed Question: RO Question # 66

The plant is operating in MODE 1 at 100% power with the following conditions:

- Main Condenser backpressure is rising
- AOP 691, Condenser High Backpressure has been entered
- Actions are being taken IAW the AOP, including a fast power reduction to 27 Mlbm/hr flow IAW IPOI-4

IAW OP-AA-100-1000, Conduct of Operations;

(1) The CRS __(1)__ Transient Annunciator Response

(2) Alarm response procedures for those alarms associated with the resulting transient __(2)__

A. (1) must verbally authorize
(2) must be immediately referenced

B. (1) must verbally authorize
(2) may be referenced as conditions permit

C. (1) does NOT have to verbally authorize
(2) must be immediately referenced

D. (1) does NOT have to verbally authorize
(2) may be referenced as conditions permit

Explanation (Optional):

A. Incorrect – AOP entry authorizes entry into Transient Annunciator Response. Alarm Response procedures shall be referenced for all alarms except when any of the following apply: Transient annunciator response is in effect and higher priority tasks are being performed.

B. Incorrect – AOP entry authorizes entry into Transient Annunciator Response. Alarm Response procedures shall be referenced for all alarms except when any of the following apply: Transient annunciator response is in effect and higher priority tasks are being performed.

C. Incorrect – AOP entry authorizes entry into Transient Annunciator Response. Alarm Response procedures shall be referenced for all alarms except when any of the following apply: Transient annunciator response is in effect and higher priority tasks are being performed.

D. Correct – AOP entry automatically authorizes Transient Annunciator Response. From the OP-AA-10-1000, Alarm Response procedures shall be referenced for all alarms except when any of the following apply: Transient annunciator response is in effect and higher priority tasks are being performed.

Technical Reference(s): OP-AA-10-1000, attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 67

In accordance with OP-AA-101, Clearance and Tagging, which one of the following conditions would require double valve protection?

Any system where the isolated portion of the system contains …

A. conditions equal to or greater than 200 psig or 500°F.

B. conditions equal to or greater than 500 psig or 200°F.

C. radioactive concentrations in excess of 10CFR20 Appendix C limits and/or temperatures equal to or greater than 212°F

D. radioactive concentrations in excess of 10CFR20 Appendix E limits and/or temperatures equal to or greater than 212°F.

Explanation (Optional):

A. Incorrect - The values are greater than 500 psig or 200°F.

B. Correct - When isolating high energy systems (>500 psi or >200°F on piping >3/8" diameter) or hazardous chemical systems (as determined by the Safety Department or indicated in the MSDS information), then double valve isolation SHALL be used (two valves in series) when available or practical.

C. Incorrect - The values are greater than 200°F and there are no restrictions based on radiation.

D. Incorrect - The values are greater than 200°F and there are no restrictions based on radiation.

Technical Reference(s): OP-AA-101, Att 6, pg 94 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 68

The plant is operating in MODE 1 at 100% power when the “A” Recirculation MG set trips due to an electrical fault. Due to an operator error, the RO closes the “B” Recirculation Pump Suction Valve instead of the “A” Recirculation Pump Discharge Valve.

What action must be taken?

A. Take action to insert all insertable control rods within 2 hours

B. Immediately scram the reactor and carry out IPOI 5

C. Enter LCO 3.0.3 immediately and be in MODE 2 within 9 hours

D. Enter AOP 264 immediately and re-open the “B” Recirculation Pump Suction Valve and re-start the “B” Recirculation Pump. No scram is required.

Explanation (Optional):

A. Incorrect - The shutting of the only operating RR pump suction valve will trip that RR pump. This leaves the reactor in a natural circulation mode, which is prohibited by Tech Specs., and requires an immediate scram.

B. Correct - The shutting of the only operating RR pump suction valve will trip that RR pump. This leaves the reactor in a natural circulation mode, which is prohibited by Tech Specs., and requires an immediate scram.

C. Incorrect - The shutting of the only operating RR pump suction valve will trip that RR pump. This leaves the reactor in a natural circulation mode, which is prohibited by Tech Specs., and requires an immediate scram.

D. Incorrect - The shutting of the only operating RR pump suction valve will trip that RR pump. This leaves the reactor in a natural circulation mode, which is prohibited by Tech Specs., and requires an immediate scram.

Technical Reference(s): SD 264 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 69

Which one of the following is NOT an approved method of deviating from the Locked Valve List?

A. Component tagout

B. An approved procedure

C. Work Control Supervisor direction

D. Operations Shift Manager direction

Explanation (Optional):

A. Incorrect - Locked valves may only be manipulated from their required position under OP-AA-101, Clearance and Tagging

B. Incorrect - Locked valves may only be manipulated from their required position under procedures that control the testing or operation of plant systems that are prepared and approved per site administrative control procedures. Examples include an OI, RFP, RWH, SPTP, or MAT.

C. Correct - There is no allowance for the Work Control Supervisor to manipulate a locked valve.

D. Incorrect - The Operations Shift Manager may direct repositioning of a locked valve under emergency conditions or as needed to protect the health and safety of the public.

Technical Reference(s): ACP-1410.9, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC #20496

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 70

The #1 Traversing In-Core Probe (TIP) detector is stuck in the core, all other TIP detectors are in their shields. An Operator and Health Physics Technician must enter the TIP Room to verify the position of the TIP takeup reel.

In accordance with OI-878.6, Traversing In-Core Probe System, and HPP 3104.01, Control of Access to High Radiation Areas and Above, which one of the following is required?

Prior to entry into the TIP Shield area the ...

A. TIP machines shall be tagged out and the Operations Manager must sign on the tagout.

B. TIP machines shall be tagged out and the Health Physics Supervisor or designee must sign on the tagout.

C. Health Physics Supervisor shall discuss the work plans and exposure control plans with the CRS and Operator.

D. CRS shall discuss the work plans and exposure control plans with the Health Physics Technician and Operator.

Explanation (Optional):

A. Incorrect - There is no requirement for the Ops Suprv to sign on the tagout.

B. Correct - In accordance with HPP 3104.01 Control of Access to High Radiation Areas and Above, entries into the TIP Shield area and/or for entries to work on the TIP machine that would have the potential to draw the TIP into the TIP machine, the TIP machines shall be tagged out and the Health Physics Supervisor or designee shall be required to sign on the tagout.

C. Incorrect - A briefing is required if the TIP can NOT be tagged out. When the work to be performed prevents the machines from being tagged out, the Health Physics Technician providing coverage for work in the area will discuss work plans and exposure control plans with the CRS and Health Physics Supervisor.

D. Incorrect - A briefing is required if the TIP can NOT be tagged out. When the work to be performed prevents the machines from being tagged out, the Health Physics Technician providing coverage for work in the area will discuss work plans and exposure control plans with the CRS and Health Physics Supervisor.

Technical Reference(s): OI-878.6, pg 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 71

ACP-1411.25, Planned Special Exposures permits a worker who has critical skills and that is necessary for a particular job can be authorized to receive an exposure in ADDITION to the routine occupational exposure limit.

The workers Annual (TEDE) Exposure Limited can be raised to ___(1)___ if authorized by the ___(2)___.

A. (1) 5 Rem
(2) Plant Manager, Nuclear

B. (1) 10 Rem
(2) Plant Manager, Nuclear

C. (1) 5 Rem
(2) Manager, Radiation Protection

D. (1) 10 Rem
(2) Manager, Radiation Protection

Explanation (Optional):

A. Correct - The individual(s) receiving a PSE are limited to the following dose from all PSEs in one year, 5 Rems TEDE. The Plant Manager, Nuclear is responsible for the authorization of a PSE

B. Incorrect - The individual(s) receiving a PSE are limited to the following dose from all PSEs in one year, 5 Rems TEDE.

C. Incorrect - The Plant Manager, Nuclear is responsible for the authorization of a PSE

D. Incorrect - The individual(s) receiving a PSE are limited to the following dose from all PSEs in one year, 5 Rems TEDE. The Plant Manager, Nuclear is responsible for the authorization of a PSE

Technical Reference(s): ACP-1411.25, pgs 4 & 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 72

The plant is in normal full power operation on a typical workday.

The NRC has just called on the ENS phone to inform the DAEC of a confirmed terrorist attack with an explosives filled aircraft at the Brunswick plant in North Carolina.

The FAA has grounded all aircraft nationally. However, they are watching two small planes headed towards the Cedar Rapids area from the North West that have not yet responded to radio communications. Time to the site is 40 minutes.

In accordance with AOP 914 “Security Events” which operator actions if any are appropriate at this time?

A. Reduce core flow, manually scram the reactor, and evacuate the site.

B. Commence a rapid downpower of the reactor using IPOI 4, Fast Power Reduction.

C. Remain at full power, back out of any STPs that are in progress and verify all ECCS operable.

D. Remain at full power, increase plant monitoring, and take NO further actions until a plane is within 30 minutes of the site.

Explanation (Optional):

A. Incorrect - This action would be correct for a Airborne Attack Probable (in the next 30 minutes).

B. Incorrect - Per Tab 3, the plant may remain at full power.

C. Correct - The event described is an Attack on US Soil and meets the definition of an "informational airborne attack. Actions are from Tab 3.

D. Incorrect - Per AOP 914, the plant may remain at full power however many preliminary actions must be taken, including backing out of any STPs that are in progress and verify all ECCS operable.

Technical Reference(s): AOP 914, Tab 3, pg 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC #10044

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 73

The plant is operating in MODE 1 at 93% power with the following conditions:

- The Torus developed an unisolable leak
- EOP-1, RPV Control, was entered after the scram due to RPV level shrink
- RPV level has since been restored to 190 inches
- RPV pressure is 920 psig and being controlled by EHC Pressure Set

The CRS directs the RO to perform SEP 307, Rapid Depressurization with Bypass Valves, to anticipate Emergency Depressurization due to Torus Level continuing to decrease uncontrollably.

(1) Is this an appropriate action at this time?

Assume the SEP 307 actions were NOT taken as above, when the CRS directs Emergency Depressurization for this event, only 1 SRV would open. The CRS then directs the BOP to perform SEP 307 as an Alternate Depressurization System.

(2) Is this an appropriate action at this time?

A. 1) Yes
2) Yes

B. 1) Yes
2) No

C. 1) No
2) Yes

D. 1) No
2) No

Explanation (Optional):

A. Correct - SEP 307 Purpose identifies its use for when ED is anticipated and for when less than the minimum number of SRVs has opened during ED. This SEP may not be used to anticipate ED during ALC or ATWS transients, so there are times when it would not be appropriate

B. Incorrect - Listed as a Table 8 Alternate Depressurization System. As long as the MSIVs remain open, this SEP is appropriate. Selected if it is believed that all alternate systems go to the Torus

C. Incorrect - SEP would not be appropriate before ED for two other types of transients, but would be for this one

D. Incorrect - SEP would not be appropriate before ED for two other types of transients, but would be for this one. Listed as a Table 8 Alternate Depressurization System. As long as the MSIVs remain open, this SEP is appropriate. Selected if it is believed that all alternate systems go to the Torus

Technical Reference(s): SEP 307
EOP Bases, EOP-1 Page 34 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 96.06.06.06
95.00.00.20 (As available)

Question Source: Bank # 2005 NRC #74

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Proposed Question: RO Question # 74

The plant is operating in MODE 1 at 100% power with the following conditions:

- A loss of Startup Transformer 1X3 and Aux Transformer 1X2 occurred
- The reactor automatically scrammed
- EOP 1, RPV Control, was entered due to RPV level shrink on the scram
- IPOI 5, Reactor Scram, has been entered
- AOP 304.1, Loss of 4160 VAC Non Essential Power, has been entered

Two minutes later Torus Water Temperature is 95°F and rising.

Which of the following actions is required?

A. Concurrently enter EOP-2, PRIMARY CONTAINMENT CONTROL

B. Continue IPOI-5, REACTOR SCRAM and monitor Torus Water Temperature, entry into EOP 2 is not required

C. Exit BOTH IPOI-5 and AOP-304.1 and enter EOP-2, PRIMARY CONTAINMENT CONTROL

D. Exit IPOI-5, REACTOR SCRAM, and enter EOP-2, PRIMARY CONTAINMENT CONTROL

Explanation (Optional):

A. Correct - with Torus Water Temperature above 95 ☐F, it is required to concurrently enter EOP-2, Primary Containment Control

B. Incorrect - would be true below 95☐F Torus Water Temperature

C. Incorrect - would be true after actions of BOTH IPOI-5 and AOP-304.1 are complete

D. Incorrect - would be true if IPOI-5 actions were complete prior to exceeding 95☐F Torus Water Temperature

Technical Reference(s): EOP-2 entry condition (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 11260

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments

4-4-11 – is there a procedure usage reference we could use?

Proposed Question: RO Question # 75

The plant was shutdown fourteen days ago for a refueling outage, with maintenance occurring that has the potential to drain the reactor vessel (OPDRV).

An Operator contacts the Control Room and informs you that someone has blocked open both reactor building airlock doors.

Which one of the following actions is required?

A. Within four hours verify one airlock door closed or stop maintenance with the potential to drain the reactor vessel (OPDRV)

B. Immediately stop maintenance with the potential to drain the reactor vessel (OPDRV) while initiating action to close at least one air lock door

C. Within four hours verify one airlock door closed or stop any refueling activities on the Refuel Floor, maintenance with the potential to drain the reactor vessel (OPDRV) may continue

D. Immediately stop any refueling activities on the Refuel Floor and initiate action to close at least one air lock door, maintenance with the potential to drain the reactor vessel (OPDRV) may continue

Explanation (Optional):

A. Incorrect - immediate actions is required by T.S.

B. Correct - With Secondary Containment inoperable initiate actions to suspend OPDRVs.

C. Incorrect - Immediately and maintenance with the potential to drain the reactor vessel (OPDRV) must be stopped

D. Incorrect - Immediately and maintenance with the potential to drain the reactor vessel (OPDRV) must be stopped

Technical Reference(s): T.S. 3.6.4.1.C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Hope Creek

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Proposed Question: SRO Question # 76

During an accident the following plant conditions exist:

- RPV pressure is 440 psig and lowering
- RPV water level is 40 inches and lowering
- Drywell pressure is 4 psig and rising
- Drywell temperature is 160°F and rising
- Torus water level is 8.9 ft and stable
- Torus pressure is 3.9 psig and rising
- All RHR pumps are running
- HPCI and RCIC are injecting
- All Control Rods are Full In

Which one of the following is required to be directed based upon the above conditions?

A. IAW EOP-1, Enter EOP-ED and Emergency Depressurize using the ADS SRVs.

B. IAW EOP-2, Primary Containment Control, initiate Drywell Spray.

C. IAW OI 151, Core Spray, attempt to raise Torus level.

D. IAW EOP-1, Anticipate ED and open Bypass Valves.

Explanation (Optional):

A. Incorrect – RPV level has not yet lowered per the EOPs to require ED. No other parameters dictate performing an ED.

B. Incorrect – With RPV level lowering, initiating Torus Spray is not performed unless adequate core cooling is assured.

C. Correct – IAW EOP 2 step T/L-1 with Torus level < 10.1 " then attempt to raise level using HPCI or Core Spray. (HPCI is currently injecting so Core Spray may be used

D. Incorrect – Anticipating ED is not permitted with an RPV level issue.

Technical Reference(s): EOP 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question:	SRO Question # 77

The plant is operating in MODE 1 at 48% power with the following conditions:

- Core Flow is 27 Mlbm/hr
- Both Recirculation Pumps are at 46% speed
- The “A” Recirculation MG Lube Oil Pressure lowers to 28 psig

15 seconds later:
(1) What is the status of the Recirc Pumps and core flow indication? AND
(2) What actions are required to be directed?

A. (1) “A” Recirc Pump is Tripped | “A” Recirc Loop has FWD Flow and indication is NOT accurate

(2) Enter OI 264, Reactor Recirc System, and direct the M/G Set Emergency DC Lube Oil Pump supply breaker 1D4202 be turned ON, and start the pump to raise lube oil pressure.

B. (1) “A” Recirc Pump is Operating | “A” Recirc Loop has FWD Flow and indication is accurate

(2) Enter OI 264, Reactor Recirc System, and direct the M/G Set Emergency DC Lube Oil Pump supply breaker 1D4202 be turned ON, and start the pump to raise lube oil pressure.

C. (1) “A” Recirc Pump is Tripped | “A” Recirc Loop has FWD Flow and indication is NOT accurate

(2) Enter AOP-255.2, Power/Reactivity Abnormal Change, and evaluate Power to Flow, if in the Exclusion region and no instabilities exist, direct RO to insert Control Rods or raise Recirc Flow to exit the Region.

D. (1) “A” Recirc Pump is Operating | “A” Recirc Loop has FWD Flow and indication is accurate

(2) Enter AOP-255.2, Power/Reactivity Abnormal Change, and evaluate Power to Flow, if in the Exclusion region and no instabilities exist, direct RO to insert Control Rods or raise Recirc Flow to exit the Region.

Explanation (Optional):

A. Incorrect – Both LOOPS are operating there is no direction in the procedure to start the DC pump, it should auto start as lube oil pressure continues to lower.

B. Incorrect – the A pump has tripped due to low lube oil pressure.

C. Correct – IAW SD 264 With Lube Oil less than 30 psig for 6 seconds will cause an MG TRIP. With A RR MG Set <50% speed (45%), there will be forward flow through both Loop. With the A RR MG Breakers OPEN, Flow subtraction will cause Total Core Flow Indication to be inaccurate. It is required to raise B Recirc MG Set Speed or Insert Control Rods to exit the Exclusion Zone per AOP-255.2, Power / Reactivity Abnormal Change.

D. Correct – indicated core flow is not accurate and inserting control rods to exit the region is permitted.

Technical Reference(s): AOP 255.2 Rev .35 page 5 Caution
SD 264 Rev.12 page 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 4054
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 78

The plant is operating in MODE 1 at 100% power with the following conditions:

- GSW temperatures have started to rise
- Field operators confirm that return line temperatures for GSW loads are rising

(1) What are loads directly affected by GSW?
(2) Actions to be directed?

A. (1) Instrument and Service Air Compressors, Drywell Cooling Coils and the Isophase Bus Duct Cooler.
(2) Enter into AOP 411, GSW Abnormal Operation, and direct the reducing of Main Generator load to less than 11,000 amps
AND
Direct the NSPEO to take manual control of TC-4717, GSW FROM GENERATOR HYDROGEN COOLERS and throttle OPEN the valve.

B. (1) Steam Tunnel Cooler, Exciter Air Cooler and the Isophase Bus Duct Cooler.
(2) Enter into AOP 411, GSW Abnormal Operation, and direct the reducing of Main Generator load to less than 11,000 amps
AND
Direct the NSPEO to take manual control of TC-4717, GSW FROM GENERATOR HYDROGEN COOLERS and throttle OPEN the valve.

C. (1) Control Building Chiller, Drywell Cooling Coils and the Steam Tunnel Cooler.
(2) Enter AOP 573, Primary Containment Control, and direct venting the drywell to maintain 1.0 to 1.5 psig
AND
Enter IPOI 4, Shutdown, and direct a Fast Power Reduction.

D. (1) Steam Tunnel Cooler, Exciter Air Cooler and the Isophase Bus Duct Cooler.
(2) Enter AOP 573, Primary Containment Control, and direct venting the drywell to maintain 1.0 to 1.5 psig
AND
Enter IPOI 4, Shutdown, and direct a Fast Power Reduction.

Explanation (Optional):

A. Incorrect – Drywell Cooling Coils are Well Water loads

B. Correct – These are loads and actions referenced in AOP 411

C. Incorrect – Drywell Cooling Coils are Well Water Loads and Venting the Drywell is not require in a loss of GSW.

D. Incorrect – Venting the Drywell is not require in a loss of GSW

Technical Reference(s): AOP 411 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 79

The plant is operating in MODE 1 at 100% power with the following conditions:

- Annunciator 1C05B (D-4), REACTOR VESSEL HI PRESSURE ALARM alarms
- Analysis of PR-4563/4564, REACTOR PRESSURE/REACTOR WATER LEVEL indicates that reactor pressure has been slowly rising over the shift until the alarm setpoint was reached.

Which one of the following actions is the SRO required to direct?

A. Enter AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure and within 15 minutes lower reactor pressure below the Technical Specification Limit.

B. Enter AOP-255.2, Power/Reactivity Abnormal Change, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure and within 15 minutes lower reactor pressure below the Technical Specification Limit.

C. Enter AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.1.2 Reactivity Anomalies and within 15 minutes lower reactor power below 1912 Mwth by lowering reactor pressure.

D. Enter AOP-255.2, Power/Reactivity Abnormal Change and Technical Specifications 3.1.2 Reactivity Anomalies and within 15 minutes lower reactor power below 1912 Mwth by lowering reactor pressure.

Explanation (Optional):

A. Correct – Annunciator 1C05B, REACTOR VESSEL HI PRESSURE ALARM is an entry condition for AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure. The AOP directs lowering reactor pressure and Technical Specifications requires lowering reactor pressure below 1025 psig within 15 minutes.

B. Incorrect – There are no entry conditions for AOP-255.2, Power/Reactivity Abnormal Change. Although reactor power will rise OI-693.1, states DAEC experience shows that core thermal power may rise approximately two-thirds of one megawatt for each psig of pressure rise. Assuming a pressure rise of 25 psig (from 1015 to 1040) would only cause a power rise of ~17 Mwth, which is below the 3% power (~57 Mwth) rise needed to cause and APRM High Alarm.

C. Incorrect - Lowering power below 1912 Mwth may not lower pressure below 1025 psig which the Tech Specs require.

D. Incorrect There are no entry conditions for AOP-255.2, Power/Reactivity Abnormal Change. Although reactor power will rise OI-693.1, states DAEC experience shows that core thermal power may rise approximately two-thirds of one megawatt for each psig of pressure rise. Assuming a pressure rise of 25 psig (from 1015 to 1040) would only cause a power rise of ~17 Mwth, which is below the 3% power (~57 Mwth) rise needed to cause and APRM High Alarm. Additionally lower power below 1912 Mwth may not lower pressure below 1025 psig which the Tech Specs require.

Technical Reference(s): AOP 262, Tech Specs 3.4.10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 80

The plant is in MODE 2 with reactor pressure at 120 psig with the following conditions:

- Steam seals are in service
- Condenser backpressure at 5.5” Hg absolute
- 1K1 is tagged out for maintenance

Then an instrument Air leak develops in the heater bay and the plant conditions are now as follows:

- 1K-90 A and B are running and loaded
- 1K-90C did not start, maintenance is working on it
- Instrument and Service Air pressure is 86 psig and slowly lowering

Which of the following:
(1) Identify the procedure that is required to be entered?
(2) Identify an action that must be directed to the NSPEO?

A. (1) Enter AOP 518, Failure Of Instrument And Service Air
(2) Direct the NSPEO to place the Instrument Air Dryers IT-265A/B in FAIL SAFE mode by placing HS-3046A/B in the ON position

B. (1) Enter AOP 518, Failure Of Instrument And Service Air
(2) Direct the NSPEO to place the Instrument Air Dryers IT-265A/B in FAIL SAFE mode by placing HS-3046A/B in the OFF position

C. (1) Enter EOP 1, RPV Control
(2) Direct the NSPEO to throttle OPEN MO-4165, Offgas Jet Compressor Regulator Bypass, to raise steam flow to at least 4300 lbm/hr.

D. (1) Enter EOP 1, RPV Control
(2) Direct the NSPEO to isolate Deluge 11 (Aux Transformer) by shutting V-33-73, East Turbine Building.

Explanation (Optional):

A. Incorrect – AOP 518 would be entered for these plant conditions. The Instrument Air Dryers are placed in the Fail Safe mode by placing HS-3046A/B in OFF.

B. Correct - AOP 518 would be entered for these plant conditions. The Instrument Air Dryers are placed in the Fail Safe mode by placing HS-3046A/B in OFF.

C. Incorrect – EOP 1 is not REQUIRED to be entered at an air pressure of 86 psig. At the given plant conditions, the Offgas System is not in service.

D. Incorrect – EOP 1 is not REQUIRED to be entered at an air pressure of 86 psig. Isolate Deluge 11 is an action at 50 psig Instrument Air pressure, or if pressure is lowering rapidly.

Technical Reference(s): AOP 518 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 81

The plant is in MODE 5 with the fuel pool gates installed for a RFO with the following conditions:

- Fuel is being moved in the Spent Fuel Pool.

Then, annunciator FUEL POOL COOLING PANEL 1C-65/1C-66 TROUBLE 1C04B D-2 alarms. The following plant conditions exist:

- The cause of the alarm is Skimmer Surge Tank Low Level
- Spent Fuel Pool level is 35 feet and slowly lowering
- Fuel Pool Cooling Pump 1P-214A is running
- Refuel Floor ARMs have increased by 2 mr/hr

(1) Has the TS LCO for Spent Fuel Pool Level requirements been exceeded, if so, what actions are required?
(2) What, if any, EAL is required to be declared?

A. (1) The TS LCO is met.
(2) NO EAL has been exceeded.

B. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended immediately.
(2) An Unusual Event must be declared.

C. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended within 1 hour if level is not corrected.
(2) NO EAL has been exceeded.

D. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended within 4 hours if level is not corrected.
(2) An ALERT must be declared.

Explanation (Optional):

A. Incorrect –The LCO has been exceeded and UE is required IAW RU2.1

B. Correct – IAW ARP 1C04B A-4 Section 3.7 & 3.8. TS 3.7.8 LCO limit is 36 feet, action is required to immediately suspend fuel movement.

C. Incorrect - A UE must be declared

D. Incorrect – An ALERT level has not been exceeded

Technical Reference(s): TS 3.7.8
EAL-02 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL-01,02

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 2, 7

Facility operating limitations in the technical specifications and their bases.
Fuel handling facilities and procedures.

Comments:

Proposed Question: SRO Question # 82

The plant is in MODE 4 with the following conditions:

- RCS temperature is 100°F
- RPV level at 200 inches
- A" RHR, "A" RHRSW and "A" ESW pumps are running
- MO-1909, Shutdown Cooling Outboard Isolation valve, goes shut due to an equipment malfunction and CANNOT be opened

What actions are required to be directed?

A. Enter a Technical Specifications Action Statement and initiate a Group 3 isolation signal immediately.

B. Enter AOP 149 and direct the evacuate of all personnel from the Refuel Floor areas EXCEPT personnel assigned to monitor for leakage and increased airborne radioactivity levels.

C. Enter AOP 149 and direct the RO to raise reactor water level to between 230 and 240 inches as measured on Floodup Range of RPV level indication.

D. Enter AOP 149 and direct that the Safety/Relief Valves are used to initiate Feed and Bleed to the Torus within one hour.

Explanation (Optional):

A. Incorrect – This would be correct with rising airborne radiation levels

B. Incorrect –.A follow up action of AOP 149 is to Evacuate all personnel from the *Drywell and Torus* areas except personnel assigned to monitor for leakage and/or increased airborne radioactivity levels. This choice lists the *Refuel floor* as the evacuation area. Evacuate Refuel floor is an action in AOP 981, Fuel Handling Event.

C. Correct – IAW AOP 149 Step

D. Incorrect – The vessel head must be on and tensioned to use this method.

Technical Reference(s): AOP 149 Rev.32 Step 3.b.
TS 3.9.8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2,5

Facility operating limitations in the technical specifications and their bases.
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 83

The plant was operating in MODE 1 at 100% power when an electrical transient occurred resulting in an automatic reactor scram.

Following the scram:

- All Control Rods are Full In
- Reactor level lowered to 155 inches and recovered to 178 inches and is slowly rising
- Drywell Pressure is 1.7 psig and stable
- Drywell temperature is 120°F and stable
- Torus Water Level is 10.2' and rising slowly
- The Main Turbine is tripped
- The SBDGs are running unloaded with the Essential Buses on their normal supply

Then, another electrical transient occurs resulting in a total loss of RPS power. RPS power cannot be restored.

Which of the following is a successful pressure control strategy that you would select when directing EOP 1, RPV Control?

A. Cool down the RPV with the Main Turbine Bypass Valves, not exceeding 100°F/hr cooldown rate.

B. Place all SRV handswitches in AUTO and bypass Main Condenser High Backpressure Isolation by installing Defeat 17 to open the MSL drain valves, MO-4423 and MO-4424.

C. Place RWCU in the Recirc Mode of operation IAW SEP 302.1 OR the Drain Mode of operation IAW SEP 302.2.

D. Place RCIC or HPCI in pressure control mode. EOP Defeat 1 may be installed

Explanation (Optional):

A. Incorrect – With the MSIVs closed, the bypass valves are not available

B. Incorrect – The MSIVs are already closed so installing this Defeat would not be useful

C. Incorrect – With total loss of RPS, PCIS group 5 would not allow RWCU to be used.

D. Correct – The Group 6 isolation removes HPCI and RCIC from available use

Technical Reference(s): EOP-1 Step RC/P-4 and Table 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 84

The plant was operating in MODE 1 at 100% power with the following conditions:

- An unisolable coolant leak occurred in the Reactor Building
- RPV level lowered to the point that fuel became uncovered and fuel damage occurred
- RPV level was recovered to normal
- Offgas Kaman readings are 6E+0 µCi/cc and slowly rising
- The RED annunciator on panel 1C35 for REACTOR BLDG KAMAN 3, 4, 5 ,6 , 7,& 8 HI RAD OR MONITOR TROUBLE alarms
- The Reactor Building Exhaust Fans (1V-EF-11A & B) and the Main Plant Exhaust Fans (1V-EF-1, 2, & 3) responded as designed

Given the above:

(1) What actions must be directed to mitigate this condition?
(2) What EAL must be declared?

A. (1) Enter EOP-4, Radioactivity Release Control, and direct operators to TRIP the Main Plant Exhaust Fans.
(2) Declare an ALERT when it is determined the release will last for greater than 15 minutes.

B. (1) Enter EOP-4, Radioactivity Release Control, and direct operators to RESTART the Main Plant Exhaust Fans.
(2) Declare an ALERT when it is determined the release will last for greater than 60 minutes.

C. (1) Enter OI-734, Reactor Building HVAC, and direct operators to TRIP the Reactor Building Exhaust Fans
(2) No EAL is required.

D. (1) Enter OI-734, Reactor Building HVAC, and direct operators to RESTART the Reactor Building Exhaust Fans.
(2) No EAL is required.

Explanation (Optional):

A. Correct – At <170 inches a Group 3 isolation occurs which trips EF-11A&B, closes AD-13A & B, and aligns SBGT to draw on the RB Vent Shaft. EF1/2/3 continue to run and draw on the Main Plant Exhaust Plenum. The RB Vent Shaft and the MP Exhaust Plenum are physically separated by only a wall which, in the history of the plant, has been found to be cracked. Also the dampers AD-13A/B could be leaking, also allowing the RB Vent Shaft to flow to the MP Exhaust Plenum and out past EF-1/2/3 which normally continue to run after a Group 3 isolation. This is a real enough concern that the is a P&L in the Reactor Building HVAC OI, a Continuous Recheck statement in EOP-4 and Steps in ARP 1C35A C-3. EOP-4 is not provided.

B. Incorrect - A high concentration of activity will cause this alarm. As part of the EOP-4 Recheck statement, Operators are directed to restart the Turbine Bldg Exhaust Fans, not Main Plant Exhaust Fans which would still be running

C. Incorrect - It was given that EF-11A/B respond as designed which means that they trip on a Group 3 isolation. If they were running, drawing on the RB Vent Shaft and discharging into the MP Exhaust Plenum under EF-1/2/3, they could cause this alarm in this scenario

D. Incorrect – Selected if the RB Kaman monitors are believed to be in the RB Vent Shaft rather than on the discharge of the MP Exhaust Fans. Operators are directed to restart the Turbine Bldg Exhaust Fans, not Reactor Building Exhaust Fans

Technical Reference(s): OI-734, P&L #4; EOP-4; ARP 1C35A, C-3 ARP 1C05B C-8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SRO 6.72.01.06 Evaluate plant conditions and CR indications and determine if any step of EOP-4 should be performed (As available)

Question Source: Bank # DAEC 2005

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 85

The plant is in MODE 2 at 6% power and stable with the following plant conditions:

- RPV pressure is 780 psig and stable
- The Mechanical Vacuum pump is OFF
- Annunciator 1C34, C-3 OFFGAS JET COMPRESSOR LO STEAM FLOW alarms
- Offgas flow is lowering and is being controlled using Offgas Flow Controller HIC-4151
- CV-4151, OG JET COMPRESSOR 1S-111 SUCTION ISOLATION, is partially OPEN but will NOT open further
- Condenser vacuum is degrading and there is no indication that the loss of Offgas is due to a steam leak

Which of the following is required to be directed?

A. Enter OI 691, Condenser Air Removal, and direct the shutdown of the Condenser Air Removal System. Then direct the SJAE and Offgas MSL ‘A’ and ‘B’ steam supplies MO-1362A and MO-1362B be SHUT.

B. Enter OI 150, Reactor Core Isolation Cooling, and direct an RO to place RCIC on service in the Pressure Control Mode. When it is determined that the Offgas System cannot be restored, direct a manual scram and enter IPOI-5, Reactor Scram.

C. Enter AOP 672.1, Loss of Offgas System, and direct that CV-4151 be failed open and control pressure using MO-4151. When it is determined that the Offgas System cannot be restored, direct a manual scram and enter IPOI-5, Reactor Scram.

D. Enter AOP 672.1, Loss of Offgas System, and direct that CV-4151 be failed open and control pressure using MO-4151. When it is determined that the Offgas System cannot be restored, then enter OI 691, Main Condenser and Main Condenser Air Removal System, and direct the shutdown of the Condenser Air Removal System and the starting of the mechanical Vacuum Pump.

Explanation (Optional):

A. Incorrect – this is the action if a steam leak is the cause.

B. Incorrect – these are actions if power is >10%

C. Incorrect – these are actions if power is >10%

D. Correct – IAW AOP 672.1 Steps 10 & 13

Technical Reference(s): AOP 672.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 86

Following a small steam break in the Drywell the following conditions exist:

- Containment pressure is 4 psig, and rising very slowly
- The Crew has entered EOP-1, RPV Control and EOP-2, Primary Containment Control
- SRV manual initiations were required to cooldown the RPV
- All automatic actions occurred as expected

Given the above:
(1) What is the impact of a LONG TERM Nitrogen supply isolation? AND
(2) What actions are required to be directed?

A. (1) SRVs will fail to open for manual RPV pressure control OR for initiation of ADS.
(2) Direct the installation of EOP Defeat 11, Containment N2 Isolation Defeat, and place CV-4371A GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to ensure SRV Nitrogen Accumulators remain charged.

B. (1) SRVs will still open in response to an RPV high pressure condition OR a manual signal, but will not an open for an ADS initiation.
(2) Direct the installation of EOP Defeat 11, Containment N2 Isolation Defeat and place CV-4371A GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to directly supply nitrogen to the SRVs.

C. (1) SRVs will fail to open for manual RPV pressure control or for initiation of ADS.
(2) Direct the installation of EOP Defeat 9, Group 3, High DW Press & Rx Low Level Isolation Defeat, and place CV-4371C GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to ensure SRV Nitrogen Accumulators remain charged.

D. (1) SRVs will still open in response to an RPV high pressure condition OR a manual signal, but may not an open for an ADS initiation.
(2) Direct the installation of EOP Defeat 9, Group 3, High DW Press & Rx Low Level Isolation Defeat, and place CV-4371C GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to directly supply nitrogen to the SRVs.

Explanation (Optional):

A. Correct - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. The correct EOP Defeat is #11 which is specifically intended for this condition.

B. Incorrect - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. Without Nitrogen pressure the SRVs cannot be manually opened.

C. Incorrect - EOP Defeat 9 does NOT contain a step for bypassing the Compressor Isolation Valves. On a Group 3 isolation signal (SBGT Lockout), both of these valves will close. CV-4371A has a 2 position keylock NORM-OVERRIDE OPEN switch (Key removable in NORM) which can be used to open the valve with a Group 3 signal present. CV-4371C has no such feature. Defeat 9 is intended for defeating the High Drywell Pressure and RPV Low Water Level Isolations from Group 3 PCIS logic. This enables restoration of Reactor Building ventilation and/or containment vent valves.

D. Incorrect - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. Without Nitrogen pressure the SRVs cannot be manually opened. EOP Defeat 9 does NOT contain a step for bypassing the Compressor Isolation Valves. On a Group 3 isolation signal (SBGT Lockout), both of these valves will close. CV-4371A has a 2 position keylock NORM-OVERRIDE OPEN switch (Key removable in NORM) which can be used to open the valve with a Group 3 signal present. CV-4371C has no such feature. Defeat 9 is intended for defeating the High Drywell Pressure and RPV Low Water Level Isolations from Group 3 PCIS logic. This enables restoration of Reactor Building ventilation and/or containment vent valves.

Technical Reference(s): SD-183.1
SD 573, pg 25
EOP Defeat 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

12-16-10, The second part of responses C and D could be (2) At Panel 1C15 and C-17, place GROUP 3 CHANNEL A HI DW AND RX LO LEVEL OVERRIDE keylock switches in OVERRIDE. I thought going with the wrong valve and a step that is not even in Defeat 9 made the answers more wrong and gave the candidates a better shot at the question. (do you have any opinion on this distracter?)

Proposed Question: SRO Question # 87

The plant is operating in MODE 1 at 50% power with the following conditions:

- The Feedwater Level Control System is in THREE element control
- FT-1581, the "A" Feedwater Flow Transmitter outputs fails to zero (0)

What will be:
(1) What procedure you would enter, and required actions to be directed? AND
(2) What secondary actions you will direct?

A. (1) Direct entry into OI-644, Feedwater/Condensate, Section 4.3, Selecting Reactor Water Level Control Input, and direct the RO to shift feedwater control to single element.
(2) No other actions necessary, the circuit will correct RPV level transient with no further actions.

B. (1) Direct entry into AOP-644, Feedwater/Condensate Malfunction, direct the RO to take manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.
(2) Direct the RO to throttle the FRV's CLOSED to recover RPV water level, then stabilize the level in the green band.

C. (1) Direct entry into OI-644, Feedwater/Condensate, Section 4.3, Selecting Reactor Water Level Control Input, and direct the RO to shift feedwater control to single element.
(2) Direct the RO to throttle the FRV's OPEN to recover RPV water level, then stabilize the level in the green band.

D. (1) Direct entry into AOP-644, Feedwater/Condensate Malfunction, and direct the RO to take manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.
(2) No other actions necessary, the circuit will correct RPV level transient with no further actions.

Explanation (Optional):

A. Incorrect - This section of OI-644 does not specify placing feedwater in single element control. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.

B. Correct - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.

C. Incorrect - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow. This section of OI-644 does not specify placing feedwater in single element control. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.

D. Incorrect - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow.

Technical Reference(s): AOP-644, SD-644 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 88

The reactor was scrammed due to an un-isolable Feedwater line break outside the Primary Containment. Busses 1A1 and 1A2 tripped and locked out due to the feedwater leak. Current plant conditions are as follows:

- RPV water level is 100 inches, rising slowly
- HPCI and RCIC are injecting
- Drywell pressure is 2.7 psig, stable
- Torus water level is 7.1 feet, lowering at 0.1 feet per minute
- Torus water temperature is 88°F, rising slowly

Which one of the following actions:
(1) What procedure is required to be entered and what actions are required to be taken?
(2) What is the bases for this action?

A. (1) Enter RCIC Quick Response Card (QRC) and HPCI QRC, and direct the RO to TRIP HPCI and RCIC
(2) Prevent direct pressurization of the Primary Containment

B. (1) Enter SEP 307, and direct to RO to open turbine Bypass Valves in anticipation of Emergency Depressurization
(2) Minimize heat addition to the torus

C. (1) Enter the Emergency Depressurization EOP and direct the RO to open 4 ADS SRVs to emergency depressurize the RPV
(2) Prevent exceeding the HCTL curve

D. (1) Enter the Emergency Depressurization EOP and direct the RO to open 4 ADS SRVs to emergency depressurize the RPV
(2) Due to uncovering the downcomers

Explanation (Optional):

A. Incorrect – HPCI is required to be secured at 5.8 ft torus level, not 7.1 ft

B. Incorrect – EOP 2 requires ED BEFORE torus level reaches 7.1 ft

C. Incorrect - Torus level of 7.1 ft corresponds to a loss of PSP of the containment, not exceeding the HCTL curve.

D. Correct - A torus level of 7.1 ft. corresponds to the bottom of the drywell-to-torus downcomers. Torus levels below 7.1 ft. would result in loss of the pressure suppression function of the primary containment (e.g., during a LOCA, steam entering the torus would not be fully condensed).

Technical Reference(s): EOP-2 Bases, pg 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question:	SRO Question # 89

The plant is operating in MODE 1 at 100% power with the following conditions:

- Several annunciators alarm on the 1C08 panel
- Fifteen minutes later the following annunciators are still in their alarm condition:
  - 1C08A, A-5, BUS 1A3 LOCKOUT TRIP
  - 1C08B, A-6, BUS 1A4 LOCKOUT TRIP
  - 1C08B, A-3, S/U XFMR TO 1A1 BREAKER 1A102 TRIP
  - 1C08A, A-8, S/U XFMR TO 1A2 BREAKER 1A202 TRIP

As the CRS you must direct the crew to verify the…

A. Standby Diesel Generators are shutdown and declare a SITE AREA EMERGENCY based on a loss of all offsite power and loss of all onsite AC power to Essential Buses for 15 minutes.

B. Standby Diesel Generators are shutdown and declare an ALERT based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

C. Standby Diesel Generators are supplying Buses 1A3 and 1A4 and declare a SITE AREA EMERGENCY based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

D. Standby Transformer is supplying Buses 1A2, and either 1A3 or 1A4 and declare a ALERT based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Explanation (Optional):

A. Correct – With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses. The SBDG will start but will not have cooling water and the ARPs for the respective lockout annunciators direct shutting down the SBDGs. With Breakers 1A101 and 1A201 tripped no power is available to buses 1A1 and 1A2 from either the Startup or Standby Transformers. Because the plant tripped on a loss of RPS there is no power available from the Auxiliary Transformer. Therefore the station is in a Blackout condition and has been for 15 minutes. IAW EAL-01, a SITE AREA EMERGENCY based on a loss of all offsite power and loss of all onsite AC power to Essential Buses for 15 minutes.

B. Incorrect - With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses.

C. Incorrect - With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses.

D. Incorrect - With Breakers 1A101 and 1A201 open no power is available to buses 1A1 and 1A2 from either the Startup or Standby Transformers.

Technical Reference(s): ARP 1C08A, A-5 & A-8; ARP 1C08B, A-3 & A-6; EAL-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL-1

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 90

The plant is operating in MODE 1 at 60% power with the following conditions:

- Main Condenser Backpressure is 4 inches Hg absolute, slowly rising
- Blue SCRAM lights are ON for Control Rods 38-31 and 42-27
- 1C05A D-6 ROD DRIFT alarms
- 1C05B F-1 SCRAM AIR HEADER HI/LO PRESSURE alarms
- 1C05A E- 3 SBLC TANK HI/LO LEVEL alarms
- Condensate Recirculation Flow Control Valve CV-1428 is SHUT

Which one of the following describes:
(1) The expected plant condition, and
(2) The appropriate mitigating procedure?

Instrument Air Header Pressure is:

A. (1) below 70 psig.
(2) It is required to enter and direct actions per AOP-518, Failure of Instrument and Service Air. A Reactor scram must be directed.

B. (1) below 70 psig.
(2) It is required to enter and direct actions per AOP-518, Failure of Instrument and Service Air and AOP-644 Feedwater / Condensate Malfunction. A Reactor scram is not required at this time.

C. (1) above 80 psig.
(2) It is required to enter and direct actions per AOP-691, Condenser High Backpressure and IPOI-5, Reactor Scram.

D. (1) above 80 psig.
(2) It is required to enter and direct actions per AOP-255.1, Control Rod Movement/Indication Abnormal and AOP-644 Feedwater/Condensate Malfunction.

Explanation (Optional):

A. Correct - Scram Air Header Lo alarms at 68 psig, which indicates Instrument Air Header Pressure is below 80 psig. This requires AOP-518 execution. Rod Drifts require Reactor Scram.

B. Incorrect - This would be true without Rod Drift indications.

C. Incorrect - This would be true above 7.5 inches Hg without Rod Drift indications.

D. Incorrect - Rod Drift due to Instrument Air Loss is not covered by AOP-255.1. Condensate Recirculation valve failure may induce RPV Level variation.

Technical Reference(s): AOP-518 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 4071

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 91

The plant is operating in MODE 1 at 62% power with the following conditions:

- The plant is starting up from a maintenance outage
- Core Flow is 37 Mlbm/hr
- The “A” RFP develops a severe oil leak and trips
- RPV level lowers to 184 inches

Which one of the following is the:
(1) What are the impacts on the plant due to this event? AND
(2) What procedures must you enter and what action do you direct?

A. (1) The Recirculation Pumps will run back to 45% speed after a 15 second time delay
(2) Enter into ARP 1C06B C-3, “A” RX FEED PUMP 1P-1A TRIP OR MOTOR OVERLOAD, direct the RO to manually scram the reactor.

B. (1) The Recirculation Pumps will run back to 20% speed after a 15 second time delay
(2) Enter into ARP 1C06B C-3, “A” RX FEED PUMP 1P-1A TRIP OR MOTOR OVERLOAD, direct the RO to manually scram the reactor.

C. (1) The Recirculation Pumps will run back to 45% speed
(2) Enter AOP 644, Feedwater/Condensate Malfunction, and direct the RO to stabilize RPV level

D. (1) The Recirculation Pumps will run back to 20% speed
(2) Enter AOP 644, Feedwater/Condensate Malfunction, and direct the RO to stabilize RPV level

Explanation (Optional):

A. Incorrect – 45% runback has no 15 second time delay, that is on 20% runback, also, reactor scram is not required if initial reactor power is below 75% power.

B. Incorrect – 20% runback would not be activated (FW flow not <20%), also, reactor scram is not required if initial reactor power is below 75% power.

C. Correct - A 45% runback would be initiated, since only 1 RFP pump is running, with RPV level < 185”. A scram is not required since starting power is 62, AOP 644 would be entered and level stabilized.

D. Incorrect - 20% runback would not be activated (FW flow not <20%),

Technical Reference(s): ARP 1C06B, C-3
AOP 644 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: RO Question # 92

The plant was operating in MODE 1 at 100% power when an event occurred which required a manual reactor scram. Current plant conditions are as follows:

- Reactor power is 12%
- 77 control rods NOT full in
- ATWS EOP entered
- Reactor Water Level band is 15” to 87” using FW
- Neither SBLC pump is available

Boron injection using RWCU has been directed.

(1) What procedure must be directed?
(2) What actions must be taken to inject Boron?

A. (1) Defeat 14 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C82 located at RB 812' Level

B. (1) Defeat 14 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C52 located at RB 786' Level

C. (1) Defeat 15 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C82 located at RB 812' Level

D. (1) Defeat 15 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C52 located at RB 786' Level

Explanation (Optional):

A. Incorrect - NRHX High Temperature is overridden at panel 1C52.

B. Correct – Defeat 14 must be installed, and NRHX High Temp is overridden at panel 1C52.

C. Incorrect - Defeat 14 must be installed, not Defeat 15. NRHX High Temperature is overridden at panel 1C52.

D. Incorrect - Defeat 14 must be installed, not Defeat 15. NRHX High Temperature is overridden at panel 1C52.

Technical Reference(s): SEP 304
Defeat 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2, 5

Facility operating limitations in the technical specifications and their bases.

Comments:

Proposed Question:	SRO Question # 93

A plant startup is in progress.

- The 1C05 operators were withdrawing a group of rods from position 12 to position 24 by group notch withdrawal.
- All the rods in that group were at position 14, and the first rod in the group was being withdrawn again.

Then the solid-state timer malfunctioned, applying a continuous withdraw signal longer than the automatic protective circuitry would allow. After the Reactor Manual Control System responded as designed, the rod was identified at position 20. No alarms were received from nuclear instrumentation.

(1) Until the timer malfunction is reset, how will the ability of operators to move control rods be impacted?
(2) Must AOP 255.1, "Control Rod Movement/Indication Abnormal" be entered because the control rod qualifies as a "Mispositioned Control Rod"?

A. (1) Operators will NOT be able to select control rods due to a RMCS select block.
(2) AOP 255.1 must be entered because the rod has withdrawn far enough to qualify as a Mispositioned Rod

B. (1) Operators will NOT be able to select control rods due to a RMCS select block.
(2) AOP 255.1 need NOT be entered because the rod has NOT withdrawn far enough to qualify as a Mispositioned Rod

C. (1) Operators will be able to select control rods but a ROD OUT BLOCK (1C05B A-6) will prevent further withdrawals due to RMCS rod block.
(2) AOP 255.1 must be entered because the rod has withdrawn far enough to qualify as a Mispositioned Rod

D. (1) Operators will be able to select control rods but a ROD OUT BLOCK (1C05B A-6) will prevent further withdrawals due to RMCS rod block.
(2) AOP 255.1 need NOT be entered because the rod has NOT withdrawn far enough to qualify as a Mispositioned Rod

Explanation (Optional):

A. Correct - This RMCS feature protects against an unrequested continuous rod withdrawal in case the timer fails. Its protective action is to de-energize the rod Select Relays until reset. Mispositioned Rods are "two notches or greater beyond the remainder of rod group". This rod is three notches further out than the rest of the group

B. Incorrect - This RMCS feature protects against an unrequested continuous rod withdrawal in case the timer fails. Its protective action is to de-energize the rod Select Relays until reset, but the Mispositioned Rod definition excludes double notches, but not triple notches. "As described" rod movement was a triple notch

C. Incorrect - Cannot select rods and no Rod Block for this malfunction. However, the Mispositioned Rod statement is correct.

D. Incorrect - Cannot select rods and no Rod Block for this malfunction. Mispositioned Rod definition excludes double notches, but not triple notches. "As described" rod movement was a triple notch

Technical Reference(s):	SD 856.1, Rev. 4, Pages 13 &16; AOP 255.1,	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	SRO 5.02.01.06 Evaluate plant conditions and CR indications and determine if entry into AOP 255.1 is required	(As available)

Question Source:	Bank #	2005 NRC SRO #17	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2005

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 94

The reactor is in MODE 5 with core alterations in progress. What are:

(1) The required qualifications of the fuel handling SRO? AND
(2) Where must he be stationed during core alterations?

A. (1) A current DAEC SRO. He need NOT be active
(2) Must be present on the refueling bridge to directly supervise core alterations

B. (1) A current DAEC SRO. He must be active
(2) May supervise fuel movement from any location on the refuel floor

C. (1) A former SRO from a similar facility as long as an active RO is manipulating the bridge
(2) May supervise fuel movement from any location on the refuel floor

D. (1) A current DAEC SRO. He must be active
(2) Must be present on the refueling bridge to directly supervise core alterations

Explanation (Optional):

A. Incorrect – SRO needs to be an Active SRO

B. Incorrect – SRO needs to directly supervise by being on the Refueling Bridge

C. Incorrect – SRO needs to be an Active SRO

D. Correct – Per RFP 403, precaution 2.11, to be the Refueling SRO, you must be an active SRO, and directly supervise all core alterations.

Technical Reference(s): RFP 403 P&L 2.11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

Proposed Question: SRO Question # 95

In accordance with ACP 103.2, 10 CFR 50.59 SCREENING PROCESS, a 10CFR 50.59 Evaluation determines if a proposed change, test or experiment requires:

A. NRC approval prior to implementation.

B. Inspection by the NRC during the activity.

C. A Technical Specification revision after implementation.

D. Evaluation for compliance with NRC Reg Guides.

Explanation (Optional):

A. Correct: 10 CFR 50.59 establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval.

B. Incorrect: NRC determines inspection requirements.

C. Incorrect: The evaluation may determine if a Technical Specifications is required it does not however always require a Technical Specification revision, and it would not allow a TS change after the fact for a facility change. However, an UFSAR change is sometimes allowed to take up to 2 years to make, and this choice would be plausible if the candidate transferred this rule to TS.

D. Incorrect: Does not evaluate compliance with NRC Reg Guides.

Technical Reference(s): ACP-103.2, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 1

Conditions and limitations in the facility license

Comments:

Proposed Question: SRO Question # 96

The plant is operating in MODE 1 at 30% power with the following conditions:

- Plant is being shutdown for a Drywell entry to find the cause of increased floor drain leakage
- Operators were about to commence an air purge (de-inerting) of the containment when both Offgas Stack Radiation Monitors, RM-4116A&B, were declared inoperable due to a failed surveillance test.
- KAMAN 9 and 10, Offgas Stack KAMAN monitors, remain in-service and operable

Which one of the following is correct regarding the operators ability to de-inert while RM-4116A&B are not operable?

De-inerting may _____.

A. NOT begin because containment venting in this situation would be an unmonitored release.

B. NOT begin because a Group 3 isolation caused by RM-4116A&B inoperability would NOT allow containment venting.

C. begin because the Offgas KAMANs being operable satisfy ODAM and Technical Specification requirements for a release.

D. begin as long as appropriate administrative controls being maintained on the containment vent and purge valves while they are open.

Explanation (Optional):

A. Incorrect - Although the ARP states RM-4116A&B are required in modes 1, 2, and 3 (during venting and purging of the Primary Containment. However T.S. Section 3.3.6.1.L.2 permits the use of alternate instrumentation.

B. Incorrect - Offgas Stack Radiation Monitors, RM-4116A&B becoming inoperable do NOT cause a Group 3 isolation. Offgas Vent Pipe Radiation Monitors, RM-4116A&B Hi HI will cause a Group 3 isolation but the downscale/inoperative does NOT.

C. Incorrect - Offgas Stack Radiation Monitors, RM-4116A&B becoming inoperable do NOT cause a Group 3 isolation. Offgas Vent Pipe Radiation Monitors, RM-4116A&B Hi HI will cause a Group 3 isolation but the downscale/inoperative does NOT.

D. Correct - RM-4116A&B are required in modes 1, 2, and 3 (during venting and purging of the Primary Containment. However T.S. Section 3.3.6.1.L.2 permits establishing administrative control of the primary containment vent and purge valves using continuous monitoring of alternate instrumentation.

Technical Reference(s): 1C03A, A-4 & C-4
T.S. 3.3.6.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S. 3.3.6.1 including Table 3.3.6.1-1

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #21161
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Proposed Question: SRO Question # 97

The plant is operating in MODE 1 at 100% power with the following conditions:

- It is day shift
- While lowering a crate of highly radioactive material from the 5th floor, the sling breaks, causing the contents of the crate to spill out on the ground floor of the Reactor Building
- No one is injured
- Railroad Access ARM is alarming and reading 30 mR/hour

The OSM takes or directs the following actions:

- Declares a Notification of Unusual Event HU-5, based on OSM judgment.
- Sounds the Evacuation Alarm.
- Makes a Plant Page announcement for all personnel to evacuate the Reactor Building.
- Repeats the Evacuation alarm and Plant Page announcement.

Which one of the following is correct concerning the OSM's compliance with the Emergency Plan?

A. All of the OSM's actions have complied with the Emergency Plan.

B. The entire plant must be evacuated when the Evacuation Alarm is used for an EAL declaration.

C. The OSM may NOT declare an EAL based on judgment, this may only be performed by the EC when the TSC has been staffed.

D. The Evacuation Alarm is only used for EAL declarations of ALERT or greater, and may not be used for a Notification of Unusual Event.

Explanation (Optional):

A. Incorrect - For accountability purposes, if we are in an emergency classification condition, evacuation of any area of the plant requires evacuation of the entire plant.

B. Correct - IAW EPIP-1.3 The OSM/CRS shall direct sounding of the Evacuation Alarm for approximately ten (10) seconds for any event classified as an ALERT or greater, it is discretionary at the NOUE classification. However for accountability purposes, if we are in an emergency classification condition, evacuation of any area of the plant requires evacuation of the entire plant.

C. Incorrect - The OSM may declare the judgment EAL

D. Incorrect - It is discretionary at the NOUE classification.

Technical Reference(s): EPIP-1.3, pg 4
EBD-H, pg 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #21162

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 98

The plant is operating in MODE 1 at 100% power with the following conditions:

- EHC Pressure Regulator "A" is in service
- The Operator At The Controls (OATC) reports a small step change (rise) of RPV pressure
- This small pressure rise is confirmed using 1C07 indications
- Reactor power is still 100% and stable
- Reactor steam dome pressure is 1021 psig and stable

Based upon these indications:
(1) The "A" pressure regulator has failed ___(1)___.
(2) The Technical Specification action required is…

A. (1) high
(2) Declare LCO 3.2.2, MCPR, NOT met, and take action to lower reactor power to less than 21.7% power within 4 hours.

B. (1) low
(2) Declare LCO 3.2.2, MCPR, NOT met, and take action to lower reactor power to less than 21.7% power within 4 hours.

C. (1) high
(2) Declare LCO 3.4.10, Reactor Steam Dome Pressure, NOT met, and take action to be in MODE 3 within 12 hours.

D. (1) low
(2) Declare LCO 3.4.10, Reactor Steam Dome Pressure, NOT met, and take action to be in MODE 3 within 12 hours.

Explanation (Optional):

A. Incorrect – The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. When one pressure regulator is out of service, OI 693.1 directs the MCPR tech spec to be entered and power to be lowered to < 21.7% within 4 hours.

B. Correct – The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. When one pressure regulator is out of service, OI 693.1 directs the MCPR tech spec to be entered and power to be lowered to < 21.7% within 4 hours.

C. Incorrect - The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. The Steam Dome Pressure TS is > 1025 psig, which is met in this question.

D. incorrect - The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. The Steam Dome Pressure TS is > 1025 psig, which is met in this question.

Technical Reference(s): OI 693.1
TS 3.7.7 and 3.4.10
SD 693.1a
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Proposed Question: SRO Question # 99

While supervising fuel handling activities in the Spent Fuel Pool, you discover a minor typographical error in the approved Fuel Moving Plan (FMP). Per RFP 403, PERFORMANCE OF FUEL HANDLING ACTIVITIES, which of the following describes the process for correcting the error to the fuel moving plan?

A. Minor pen & ink changes to the FMP may be made by the Fuel Handling Supervisor with concurrence from the Shift Manager and Operations Manager.

B. Minor pen & ink changes to the FMP are not allowed. A revised plan must be submitted for review by Reactor Engineering, the Shift Manager, and the Operations Manager.

C. Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor and the Reactor Engineer only.

D. Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

Explanation (Optional):

A. Incorrect - The Fuel Handling Supervisor and Reactor Engineer must also approve the change.

B. Incorrect - The questions states this is a minor change, minor changes are permitted with the proper reviews.

C. Incorrect - The Shift Manager must also approve the change.

D. Correct - Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

Technical Reference(s): RFP 403, Sect. 5.1.1 e, pg 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #22624

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 6

Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments:

Proposed Question: SRO Question # 100

Which one of the following completes the note below from ACP 1408.1, Work Orders, Section 3.10, Closure:

For Safety Related Systems/Components, the Control Room Supervisor (CRS) reviewing and performing the operability testing _______________ CRS who planned the operability testing in WO instructions.

A. is allowed to be the same

B. shall be different than the

C. shall review the post maintenance testing with the

D. is allowed to change that testing by verbally informing the

Explanation (Optional):

A. Incorrect - The Operations Shift Supervisor reviewing and performing the operability testing shall be different than the CRS who planned the operability testing.

B. Correct - IAW ACP-1408.1, For Safety Class SR Systems/Components, the Operations Shift Supervisor reviewing and performing the operability testing shall be different than the CRS who planned the operability testing in WO instructions.

C. Incorrect - There is no requirement for the OSS to review the test with the CRS who planned the test.

D. Incorrect - If testing requirements are to be changed, the CRS must obtain concurrence, as appropriate, from affected departments.

Technical Reference(s): ACP 1408.1, pg 52 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC #19818

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 3

Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Duane Arnold Energy Center – EAL-01 Emergency Action Level Matrix, Rev. 8 (Modes 1, 2, 3)

Sheet 1

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Abnormal Rad Release / Rad Effluent	Offsite Rad Conditions	**RG1 Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mRem TEDE or 5000 mRem CDE Thyroid for the Actual or Projected Duration of the Release Using Actual Meteorology** ☐ **RG1.1** 1 2 3 4 5 DEF Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary. (Preferred method) ☐ **RG1.2** 1 2 3 4 5 DEF If Dose Assessment is unavailable, either of the following: - Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading GREATER THAN 6 E-1 µCi/cc and is expected to continue for 15 minutes or longer. - Valid Offgas Stack rad monitor (Kaman 9/10) reading GREATER THAN 4 E+2 µCi/cc and is expected to continue for 15 minutes or longer ☐ **RG1.3** 1 2 3 4 5 DEF Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour at or beyond the site boundary; or analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation at or beyond the site boundary	**RS1 Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem CDE Thyroid for the Actual or Projected Duration of the Release** ☐ **RS1.1** 1 2 3 4 5 DEF Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary. (Preferred method) ☐ **RS1.2** 1 2 3 4 5 DEF If Dose Assessment is unavailable, any of the following: - Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading GREATER THAN 6 E-2 µCi/cc and is expected to continue for 15 minutes or longer. - Valid Offgas Stack rad monitor (Kaman 9/10) reading GREATER THAN 4 E+1 µCi/cc and is expected to continue for 15 minutes or longer ☐ **RS1.3** 1 2 3 4 5 DEF Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour at or beyond the site boundary; or analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation at or beyond the site boundary	**RA1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200X the Offsite Dose Assessment Manual (ODAM) Limit and is Expected to Continue for 15 Minutes or Longer** ☐ **RA1.1** 1 2 3 4 5 DEF Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading that exceeds 3 E-2 µCi/cc and is expected to continue for 15 minutes or longer ☐ **RA1.2** 1 2 3 4 5 DEF Valid Offgas Stack rad monitor (Kaman 9/10) reading that exceeds 6 E+0 µCi/cc and is expected to continue for 15 minutes or longer ☐ **RA1.3** 1 2 3 4 5 DEF Valid LLRPSF rad monitor (Kaman 12) reading that exceeds 1 E-1 µCi/cc and is expected to continue for 15 minutes or longer ☐ **RA1.4** 1 2 3 4 5 DEF Valid GSW rad monitor (RIS-4767) reading that exceeds 3E+5 CPS and is expected to continue for 15 minutes or longer ☐ **RA1.5** 1 2 3 4 5 DEF Valid RHRSW & ESW rad monitor (RM-1997) reading that exceeds 8E+4 CPS and is expected to continue for 15 minutes or longer ☐ **RA1.6** 1 2 3 4 5 DEF Valid RHRSW & ESW Rupture Disc rad monitor (RM-4268) reading that exceeds 1E+5 CPS and is expected to continue for 15 minutes or longer ☐ **RA1.7** 1 2 3 4 5 DEF Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates with a release duration of 15 minutes or longer in excess of 200 times ODAM limit	**RU1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Assessment Manual (ODAM) Limit and is Expected to Continue for 60 Minutes or Longer** ☐ **RU1.1** 1 2 3 4 5 DEF Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading that exceeds 1 E-3 µCi/cc and is expected to continue for 60 minutes or longer ☐ **RU1.2** 1 2 3 4 5 DEF Valid Offgas Stack rad monitor (Kaman 9/10) reading that exceeds 2.0 E-1 µCi/cc and is expected to continue for 60 minutes or longer ☐ **RU1.3** 1 2 3 4 5 DEF Valid LLRPSF rad monitor (Kaman 12) reading that exceeds 1.0 E-3 µCi/cc and is expected to continue for 60 minutes or longer ☐ **RU1.4** 1 2 3 4 5 DEF Valid GSW rad monitor (RIS-4767) reading that exceeds 3E+3 CPS and is expected to continue for 60 minutes or longer ☐ **RU1.5** 1 2 3 4 5 DEF Valid RHRSW & ESW rad monitor (RM-1997) reading that exceeds 8E+2 CPS and is expected to continue for 60 minutes or longer ☐ **RU1.6** 1 2 3 4 5 DEF Valid RHRSW & ESW Rupture Disc rad monitor (RM-4268) reading that exceeds 1E+3 CPS and is expected to continue for 60 minutes or longer ☐ **RU1.7** 1 2 3 4 5 DEF Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates in excess of 2 times ODAM limits that continue for 60 minutes or longer
	Onsite Rad Conditions	None	None	**RA2 Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel** ☐ **RA2.1** 1 2 3 4 5 DEF Report of any of the following: - Valid ARM Hi Rad alarm for the Refueling Floor North End (RM 9163), Refueling Floor South End (RM 9164), New Fuel Storage Area (RM 9178). - Valid Refueling Floor North End (RM-9163), Refueling Floor South End (RM-9164), or New Fuel Storage Area (RM-9153) ARM Reading GREATER THAN 10 mRem/hr - Valid Spent Fuel Storage Area (RM-9178) ARM Reading GREATER THAN 100 mRem/hr ☐ **RA2.2** 1 2 3 4 5 DEF Valid water level reading LESS THAN 450 inches as indicated on LI-4541 (floodup) for the Reactor Refueling Cavity that will result in Irradiated Fuel uncovering ☐ **RA2.3** 1 2 3 4 5 DEF Valid Fuel Pool water level indication (LI-3413) LESS THAN 16 feet that will result in Irradiated Fuel uncovering **RA3 Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown** ☐ **RA3.1** 1 2 3 4 5 DEF Valid area radiation levels GREATER THAN 15 mRem/hr in any of the following areas: - Control Room (RM 9162) - Central Alarm Station (by survey) - Secondary Alarm Station (by survey) ☐ **RA3.2** 1 2 3 4 5 DEF Valid area radiation monitor (RE-9168), reading GREATER THAN 500 mRem/hr affecting the Remote Shutdown Panel, 1C388	**RU2 Unexpected Increase in Plant Radiation** ☐ **RU2.1** 1 2 3 4 5 DEF RU2.1 Unplanned valid Refuel Floor ARM reading increase with an uncontrolled loss of reactor cavity, fuel pool, or fuel transfer canal water level with all irradiated fuel assemblies remaining covered by water as indicated by any of the following: - Report to control room - Valid fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering - Valid WR GEMAC Floodup indication (LI-4541) coming on scale ☐ **RU2.2** 1 2 3 4 5 DEF Any unplanned ARM reading offscale high or GREATER THAN 1000 times normal* reading *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value
Hazards	Natural & Destructive Phenomenon	(see Safe Shutdown/Vital Areas and Systems of Concern tables below)		**HA1 Natural and Destructive Phenomena Affecting the Plant Vital Area** ☐ **HA1.1** 1 2 3 4 5 DEF Receipt of the Amber Operating Basis Earthquake Light and the wailing seismic alarm on 1C35 (+/- 0.06 gravity) ☐ **HA1.2** 1 2 3 4 5 DEF Report of Tornado or high winds greater than 95MPH within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern ☐ **HA1.3** 1 2 3 4 5 DEF Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern ☐ **HA1.4** 1 2 3 4 5 DEF Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of a Safe Shutdown/Vital Area ☐ **HA1.5** 1 2 3 4 5 DEF River level ABOVE 767 feet ☐ **HA1.6** 1 2 3 4 5 DEF Uncontrolled flooding in a Safe Shutdown/Vital Area that results in degraded safety system performance as indicated in the Control Room or that creates an industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment ☐ **HA1.7** 1 2 3 4 5 DEF River level BELOW 724 feet 6 inches ☐ **HA1.8** 1 2 3 4 5 DEF Report to control room of VISIBLE DAMAGE affecting a Safe Shutdown/Vital Area	**HU1 Natural and Destructive Phenomena Affecting the Protected Area** ☐ **HU1.1** 1 2 3 4 5 DEF Earthquake detected per AOP 901, Earthquake ☐ **HU1.2** 1 2 3 4 5 DEF Report of a tornado touching down within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern ☐ **HU1.3** 1 2 3 4 5 DEF Report of winds greater than 95 mph within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern ☐ **HU1.4** 1 2 3 4 5 DEF Vehicle crash into plant structures or systems within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern ☐ **HU1.5** 1 2 3 4 5 DEF Report of an unanticipated explosion within the Plant Protected Area resulting in visible damage to permanent structures or equipment ☐ **HU1.6** 1 2 3 4 5 DEF Report of turbine failure resulting in casing penetration or damage to turbine or generator seals ☐ **HU1.7** 1 2 3 4 5 DEF River level ABOVE 757 feet ☐ **HU1.8** 1 2 3 4 5 DEF Uncontrolled flooding in a Safe Shutdown/Vital Area that has the potential to affect safety related equipment needed for the current operating mode ☐ **HU1.9** 1 2 3 4 5 DEF River level BELOW 725 feet 6 inches
	Fire or Explosion	None	None	**HA2 Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown** ☐ **HA2.1** 1 2 3 4 5 DEF Fire or explosion in any Safe Shutdown/Vital Area **AND** Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area	**HU2 Fire Within Protected Area Boundary Not Extinguished Within 15 Minutes of Detection** ☐ **HU2.1** 1 2 3 4 5 DEF Fire in buildings or areas contiguous to any Safe Shutdown/Vital Area not extinguished within 15 minutes of control room notification or verification of a control room alarm
	Toxic and Flammable Gas	None	None	**HA3 Release of Toxic or Flammable Gases Within or Contiguous to a Vital Area Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown** ☐ **HA3.1** 1 2 3 4 5 DEF Report or detection of toxic gases within or contiguous to a Safe Shutdown/Vital Area in concentrations that may result in an atmosphere Immediately Dangerous to Life and Health (IDLH) ☐ **HA3.2** 1 2 3 4 5 DEF Report or detection of gases in concentration greater than the Lower Flammability Limit within or contiguous to a Safe Shutdown/Vital Area	**HU3 Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant** ☐ **HU3.1** 1 2 3 4 5 DEF Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect normal plant operations ☐ **HU3.2** 1 2 3 4 5 DEF Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event
	Security	**HG1 HOSTILE ACTION resulting in loss of physical control of the facility** ☐ **HG1.1** 1 2 3 4 5 DEF A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of ANY of the following areas: Control Room, Remote Shutdown Panel, or the area such that operation of equipment required for safe shutdown is lost. ☐ **HG1.2** 1 2 3 4 5 DEF A HOSTILE ACTION has caused failure of Spent Fuel Pool Cooling systems and IMMINENT fuel damage is likely for a freshly off-loaded core in the pool.	**HS4 HOSTILE ACTION within the PROTECTED AREA** ☐ **HS4.1** 1 2 3 4 5 DEF A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by DAEC Security Shift Supervision.	**HA4 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat** ☐ **HA4.1** 1 2 3 4 5 DEF A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by DAEC Security Shift Supervision. ☐ **HA4.2** 1 2 3 4 5 DEF A validated notification from the NRC of an airliner attack threat less than 30 minutes away ☐ **HA4.3** 1 2 3 4 5 DEF DAEC Security Supervision reports any of the following security events that persists for 30 minutes, or greater, affecting the Plant Protected Area: – Suspicious FIRE or EXPLOSION – Significant Security System Hardware Failure – Loss of Guard Post Contact	**HU4 Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant** ☐ **HU4.1** 1 2 3 4 5 DEF A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision. ☐ **HU4.2** 1 2 3 4 5 DEF A credible site specific security threat notification. ☐ **HU4.3** 1 2 3 4 5 DEF A validated notification from NRC providing information on an aircraft threat.
	Control Room Evacuation	None	**HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established** ☐ **HS2.1** 1 2 3 4 5 DEF Control Room evacuation has been initiated **AND** Control of the plant cannot be established per AOP 915 within 20 minutes	**HA5 Control Room Evacuation Has Been Initiated** ☐ **HA5.1** 1 2 3 4 5 DEF Entry into AOP 915 for control room evacuation	None
	Emergency Director Judgment	**HG2 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency** ☐ **HG2.1** 1 2 3 4 5 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	**HS3 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency** ☐ **HS3.1** 1 2 3 4 5 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary	**HA6 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert** ☐ **HA6.1** 1 2 3 4 5 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels	**HU5 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE** ☐ **HU5.1** 1 2 3 4 5 DEF Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

Safe Shutdown/Vital Areas

Category	Area
Electrical Power	1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink / Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area, CSTs
Other	Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C55/56 Area, SBGT Room

Systems of Concern

- Reactivity Control
- Containment (Drywell/Torus)
- RHR/Core Spray/SRVs
- HPCI/RCIC
- RHRSW/River Water/ESW
- Onsite AC Power/EDGs
- Offsite AC Power
- Instrument AC
- DC Power
- Remote Shutdown Capability

Modes: 1 Power Operation · 2 Startup · 3 Hot Shutdown · 4 Cold Shutdown · 5 Refueling · DEF Defueled — **Modes 1, 2, 3**

Duane Arnold Energy Center
EAL-01 Emergency Action Level Matrix, Rev. 8

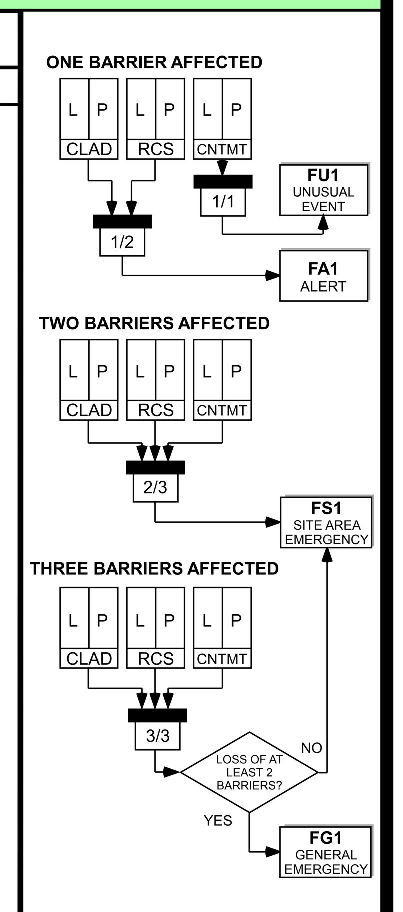
Approved: Mike Davis, Manager Emergency Preparedness — 3/25/2010 (Date)

Sheet 2

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
System Malfunct.	Loss of Power	**SG1 Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses** ☐ **SG1.1** 1 2 3 Loss of power to or from the Startup or Standby Transformer resulting in a loss of all offsite power to Emergency Busses 1A3 and 1A4 **AND** Failure of A Diesel Generator (1G-31) and B Diesel Generator (1G-21) to supply power to emergency busses 1A3 and 1A4 **AND** ANY ONE OF THE FOLLOWING: - Restoration of power to either Bus 1A3 or 1A4 is not likely within 4 hours - RPV level is indeterminate - RPV Level is LESS THAN +15 inches	**SS1 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses** ☐ **SS1.1** 1 2 3 Loss of power to or from the Startup or Standby Transformer resulting in a loss of all offsite power to Emergency Busses 1A3 and 1A4 **AND** Failure of A Diesel Generator (1G-31) and B Diesel Generator (1G-21) to supply power to emergency busses 1A3 and 1A4 **AND** Failure to restore power to at least one emergency bus, 1A3 or 1A4, within 15 minutes from the time of loss of both offsite and onsite AC power **SS3 Loss of All Vital DC Power** ☐ **SS3.1** 1 2 3 Loss of Div 1 and Div 2 125V DC busses based on bus voltage LESS THAN 105 VDC indicated for greater than 15 minutes	**SA5 AC Power Capability to Essential Busses Reduced to a Single Power Source for Greater Than 15 Minutes Such That Any Additional Single Failure Would Result in Station Blackout** ☐ **SA5.1** 1 2 3 AC power capability to 1A3 or 1A4 busses reduced to a single power source for greater than 15 minutes **AND** Any additional single failure will result in station blackout	**SU1 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes** ☐ **SU1.1** 1 2 3 Loss of power to or from the Startup or Standby Transformer resulting in a loss of all offsite power to Emergency Busses 1A3 and 1A4 that is expected to last for greater than 15 minutes **AND** Emergency Busses 1A3 and 1A4 are powered by their respective Standby Diesel Generators
	RPS Failure	**SG2 Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core** ☐ **SG2.1** 1 2 Auto Scram failure **AND** NONE of the following operator actions to reduce power are successful in shutting down the reactor: - Manual Scram Pushbuttons - Mode Switch to Shutdown - Alternate Rod Insertion (ARI) **AND** Loss of adequate core cooling or decay heat removal capability as indicated by either: - RPV level cannot be maintained GREATER THAN -25 inches - HCL Curve (EOP Graph 4) exceeded	**SS2 Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful** ☐ **SS2.1** 1 2 Auto Scram failure **AND** NONE of the following operator actions to reduce power are successful in shutting down the reactor: - Manual Scram Pushbuttons - Mode Switch to Shutdown - Alternate Rod Insertion (ARI)	**SA2 Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful** ☐ **SA2.1** 1 2 Auto Scram failure **AND** ANY of the following operator actions to reduce power are successful in shutting down the reactor: - Manual Scram Pushbuttons - Mode Switch to Shutdown - Alternate Rod Insertion (ARI)	None
	Inability to Reach or Maintain Shutdown Conditions	None	**SS4 Complete Loss of Heat Removal Capability** ☐ **SS4.1** 1 2 3 EOP Graph 4 Heat Capacity Limit is exceeded	None	**SU2 Inability to Reach Required Shutdown Within Technical Specification Limits** ☐ **SU2.1** 1 2 3 Plant is not brought to required operating mode within applicableTechnical Specifications LCO Action Statement Time
	Inst. / Comm.	None	**SS6 Inability to Monitor a Significant Transient in Progress** ☐ **SS6.1** 1 2 3 Significant transient in progress and ALL of the following: - Loss of most or all annunciators on Panels 1C03, 1C04 and 1C05. - Compensatory non-alarming indications are unavailable. - Indicators needed to monitor criticality, or core heat removal, or Fission Product Barrier status are unavailable.	**SA4 Unplanned Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators Unavailable** ☐ **SA4.1** 1 2 3 Unplanned loss of most or all 1C03, 1C04 and 1C05 annunciators or indicators associated with Safety Systems for greater than 15 minutes **AND** Either of the following conditions exist: - A significant plant transient is in progress. - Compensatory non-alarming indications are unavailable	**SU3 Unplanned Loss of Most or All Safety System Annunciation or Indication in the Control Room for Greater Than 15 Minutes** ☐ **SU3.1** 1 2 3 Unplanned loss of most or all 1C03, 1C04 and 1C05 annunciators or indicators associated with Safety Systems for greater than 15 minutes **SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities** ☐ **SU6.1** 1 2 3 Loss of ALL of the following onsite communication capabilities affecting the ability to perform routine operation: - Plant Operations Radio System - In-Plant Telephones - Plant Paging System ☐ **SU6.2** 1 2 3 Loss of ALL of the following offsite communications capability: - All telephone lines (commercial) - Microwave Phone System - FTS Phone System
	Fuel Clad Degradation	None	None	None	**SU4 Fuel Clad Degradation** ☐ **SU4.1** 1 2 3 Pretreatment Offgas System (RM-4104) Hi-Hi Radiation Alarm ☐ **SU4.2** 1 2 3 Reactor Coolant sample activity value GREATER THAN 2.0 µCi/gm dose equivalent I-131
	RCS Leakage	None	None	None	**SU5 RCS Leakage** ☐ **SU5.1** 1 2 3 Unidentified or pressure boundary leakage GREATER THAN 10 gpm ☐ **SU5.2** 1 2 3 Identified leakage GREATER THAN 25 gpm
	Inadvertent Criticality	None	None	None	**SU8 Inadvertent Criticality** ☐ **SU8.1** 3 An UNPLANNED extended positive period observed on nuclear instrumentation
ISFSI Events	Cask Confine. Boundary	None	None	None	**EU1 Damage To A Loaded Cask Confinement Boundary** ☐ **EU1.1** Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask confinement boundary: - Report by plant personnel of a tornado strike - Report by plant personnel of a seismic event ☐ **EU1.2** The following accident condition with resultant visible damage to or loss of a loaded cask confinement boundary: - A loaded transfer cask is dropped as a result of normal handling or transporting ☐ **EU1.3** Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask confinement boundary
	Security	None	None	None	None
Fission Product Barriers		☐ **FG1** 1 2 3 Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier (Table F-1)	☐ **FS1** 1 2 3 Loss or Potential Loss of ANY Two Barriers (Table F-1)	☐ **FA1** 1 2 3 ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS (Table F-1)	☐ **FU1** 1 2 3 ANY Loss or ANY Potential Loss of Containment (Table F-1)

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier		RCS Barrier		Primary Containment Barrier	
☐ Loss	☐ Potential Loss	☐ Loss	☐ Potential Loss	☐ Loss	☐ Potential Loss
☐ RADIATION/CORE DAMAGE Fuel damage assessment (PASAP 7.2) indicates at least 5% fuel clad damage **OR** Drywell Area Hi Range Rad Monitor, RIM-9184A or B reading GREATER THAN 7E+2 Rem/hr **OR** Torus Area Hi Range Rad Monitor, RIM-9185A or B reading GREATER THAN 3E+1 Rem/hr **OR** Coolant activity GREATER THAN 300 µCi/gm DOSE EQUIVALENT I-131		☐ RADIATION/CORE DAMAGE Drywell Area Hi Range Rad Monitor, RIM-9184A or B reading GREATER THAN 5 Rem/hr after reactor shutdown			☐ RADIATION/CORE DAMAGE Drywell Area Hi Range Rad Monitor, RIM-9184A or B reading GREATER THAN 3E+3 Rem/hr **OR** Torus Area Hi Range Rad Monitor, RIM-9185A or B reading GREATER THAN 1E+2 Rem/hr **OR** Fuel damage assessment (PASAP 7.2) indicates at least 20% fuel clad damage
		☐ LEAKAGE Unisolable Main Steamline Break as indicated by the failure of both MSIVs in any one line to close AND EITHER: - High MSL flow or high steam tunnel temperature annunciators - Direct report of steam release	☐ LEAKAGE RCS Leakage is GREATER THAN 50 GPM inside the drywell **OR** Unisolable primary system leakage outside the drywell as indicated by area temps or ARMs exceeding the Max Normal Limits per EOP 3, Table 6.	☐ LEAKAGE Failure of both valves in any one line to close and a downstream pathway to the environment exists **OR** Unisolable primary system leakage outside the drywell as indicated by area temps or ARMs exceeding the Max Safe Limits per EOP 3, Table 6, when Containment Isolation is required. **OR** Primary containment venting per EOPs	
☐ RPV LEVEL RPV Level LESS THAN -25 Inches	☐ RPV LEVEL RPV Level LESS THAN +15 inches	☐ RPV LEVEL RPV Level LESS THAN +15 inches			☐ RPV LEVEL Primary containment flooding required
		☐ PRIMARY CONTAINMENT ATMOSPHERE Drywell pressure GREATER THAN 2 psig and not caused by a loss of DW Cooling		☐ PRIMARY CONTAINMENT ATMOSPHERE Rapid unexplained decrease following initial increase in pressure **OR** Drywell pressure response not consistent with LOCA conditions	☐ PRIMARY CONTAINMENT ATMOSPHERE Torus pressure reaches 53 psig and increasing **OR** Drywell or Torus H_2 cannot be determined to be LESS THAN 6% and Drywell or Torus O_2 cannot be determined to be LESS THAN 5%
☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the RCS Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the RCS Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment Barrier



Modes: 1 Power Operation · 2 Startup · 3 Hot Shutdown · 4 Cold Shutdown · 5 Refueling · DEF Defueled — **Modes 1, 2, 3**

Duane Arnold Energy Center
EAL-01 Emergency Action Level Matrix, Rev. 8

Approved: Mike Davis, Manager Emergency Preparedness — 3/25/2010 (Date)

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

Abnormal Rad Release / Rad Effluent

Offsite Rad Conditions

GENERAL EMERGENCY

RG1 Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mRem TEDE or 5000 mRem CDE Thyroid for the Actual or Projected Duration of the Release Using Actual Meteorology

☐ **RG1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary. (Preferred method)

☐ **RG1.2** [1 | 2 | 3 | 4 | 5 | DEF]
If Dose Assessment is unavailable, either of the following:
- Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading GREATER THAN 6 E-1 μCi/cc and is expected to continue for 15 minutes or longer.
- Valid Offgas Stack rad monitor (Kaman 9/10) reading GREATER THAN 4 E+2 μCi/cc and is expected to continue for 15 minutes or longer

☐ **RG1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour at or beyond the site boundary; or analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation at or beyond the site boundary

SITE AREA EMERGENCY

RS1 Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem CDE Thyroid for the Actual or Projected Duration of the Release

☐ **RS1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary. (Preferred method)

☐ **RS1.2** [1 | 2 | 3 | 4 | 5 | DEF]
If Dose Assessment is unavailable, either of the following:
- Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading GREATER THAN 6 E-2 μCi/cc and is expected to continue for 15 minutes or longer.
- Valid Offgas Stack rad monitor (Kaman 9/10) reading GREATER THAN 4 E+1 μCi/cc and is expected to continue for 15 minutes or longer

☐ **RS1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour at or beyond the site boundary; or analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation at or beyond the site boundary

ALERT

RA1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200X the Offsite Dose Assessment Manual (ODAM) Limit and is Expected to Continue for 15 Minutes or Longer

☐ **RA1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading that exceeds 3 E-2 μCi/cc and is expected to continue for 15 minutes or longer

☐ **RA1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Valid Offgas Stack rad monitor (Kaman 9/10) reading that exceeds 6 E+0 μCi/cc and is expected to continue for 15 minutes or longer

☐ **RA1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Valid LLRPSF rad monitor (Kaman 12) reading that exceeds 1 E-1 μCi/cc and is expected to continue for 15 minutes or longer

☐ **RA1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Valid GSW rad monitor (RIS-4767) reading that exceeds 3E+5 CPS and is expected to continue for 15 minutes or longer

☐ **RA1.5** [1 | 2 | 3 | 4 | 5 | DEF]
Valid RHRSW & ESW rad monitor (RM-1997) reading that exceeds 8E+4 CPS and is expected to continue for 15 minutes or longer

☐ **RA1.6** [1 | 2 | 3 | 4 | 5 | DEF]
Valid RHRSW & ESW Rupture Disc rad monitor (RM-4268) reading that exceeds 1E+5 CPS and is expected to continue for 15 minutes or longer

☐ **RA1.7** [1 | 2 | 3 | 4 | 5 | DEF]
Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates with a release duration expected to continue for 15 minutes or longer in excess of 200 times ODAM limit

UNUSUAL EVENT

RU1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds Two Times the Offsite Dose Assessment Manual (ODAM) Limit and is Expected to Continue For 60 Minutes or Longer

☐ **RU1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Valid Reactor Building ventilation rad monitor (Kaman 3/4, 5/6, 7/8) or Turbine Building ventilation rad monitor (Kaman 1/2) reading that exceeds 1 E-3 μCi/cc and is expected to continue for 60 minutes or longer

☐ **RU1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Valid Offgas Stack rad monitor (Kaman 9/10) reading that exceeds 2.0 E-1 μCi/cc and is expected to continue for 60 minutes or longer

☐ **RU1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Valid LLRPSF rad monitor (Kaman 12) reading that exceeds 1.0 E-3 μCi/cc and is expected to continue for 60 minutes or longer

☐ **RU1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Valid GSW rad monitor (RIS-4767) reading that exceeds 3E+3 CPS and is expected to continue for 60 minutes or longer

☐ **RU1.5** [1 | 2 | 3 | 4 | 5 | DEF]
Valid RHRSW & ESW rad monitor (RM-1997) reading that exceeds 8E+2 CPS and is expected to continue for 60 minutes or longer

☐ **RU1.6** [1 | 2 | 3 | 4 | 5 | DEF]
Valid RHRSW & ESW Rupture Disc rad monitor (RM-4268) reading that exceeds 1E+3 CPS and is expected to continue for 60 minutes or longer

☐ **RU1.7** [1 | 2 | 3 | 4 | 5 | DEF]
Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates in excess of 2 times ODAM limits and is expected to continue for 60 minutes or longer

Onsite Rad Conditions

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

RA2 Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel

☐ **RA2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Report of any of the following:
- Valid ARM Hi Rad alarm for the Refueling Floor North End (RM 9163), Refueling Floor South End (RM 9164), New Fuel Storage (RM 9153), or Spent Fuel Storage Area (RM 9178).
- Valid Refueling Floor North End (RM-9163), Refueling Floor South End (RM-9164), or New Fuel Storage Area (RM-9153) ARM Reading GREATER THAN 10 mRem/hr
- Valid Spent Fuel Storage Area (RM-9178) ARM Reading GREATER THAN 100 mRem/hr

☐ **RA2.2** [1 | 2 | 3 | 4 | 5 | DEF]
Valid water level reading LESS THAN 450 inches as indicated on LI-4541 (floodup) for the Reactor Refueling Cavity that will result in Irradiated Fuel uncovering

☐ **RA2.3** [1 | 2 | 3 | 4 | 5 | DEF]
Valid Fuel Pool water level indication (LI-3413) LESS THAN 16 feet that will result in Irradiated Fuel uncovering

RA3 Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or to Maintain Cold Shutdown

☐ **RA3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Valid area radiation levels GREATER THAN 15 mRem/hr in any of the following areas:
- Control Room (RM 9162)
- Central Alarm Station (by survey)
- Secondary Alarm Station (by survey)

☐ **RA3.2** [1 | 2 | 3 | 4 | 5 | DEF]
Valid area radiation monitor (RE-9168), reading GREATER THAN 500 mRem/hr affecting the Remote Shutdown Panel, 1C388

UNUSUAL EVENT

RU2 Unexpected Increase in Plant Radiation

☐ **RU2.1** [1 | 2 | 3 | 4 | 5 | DEF]
RU2.1 Unplanned valid Refuel Floor ARM reading increase with an uncontrolled loss of reactor cavity, fuel pool, or fuel transfer canal water level with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:
- Report to control room
- Valid fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering
- Valid WR GEMAC Floodup indication (LI-4541) coming on scale

☐ **RU2.2** [1 | 2 | 3 | 4 | 5 | DEF]
Any unplanned ARM reading offscale high or GREATER THAN 1000 times normal* reading

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value

Hazards

Natural & Destructive Phenonenon

Safe Shutdown/Vital Areas

Category	Area
Electrical Power	1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink / Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area, CSTs
Other	Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C55/56 Area, SBGT Room

Systems of Concern
- Reactivity Control
- Containment (Drywell/Torus)
- RHR/Core Spray/SRVs
- HPCI/RCIC
- RHRSW/River Water/ESW
- Onsite AC Power/EDGs
- Offsite AC Power
- Instrument AC
- DC Power
- Remote Shutdown Capability

ALERT

HA1 Natural and Destructive Phenomena Affecting the Plant Vital Area

☐ **HA1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Receipt of the Amber Operating Basis Earthquake Light and the wailing alarm on 1C35 (+/- 0.06 gravity)

☐ **HA1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Report of Tornado or high winds greater than 95MPH within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern

☐ **HA1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern

☐ **HA1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of a Safe Shutdown/Vital Area

☐ **HA1.5** [1 | 2 | 3 | 4 | 5 | DEF]
River level ABOVE 767 feet

☐ **HA1.6** [1 | 2 | 3 | 4 | 5 | DEF]
Uncontrolled flooding in a Safe Shutdown/Vital Area that results in degraded safety system performance as indicated in the Control Room or that creates an industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment

☐ **HA1.7** [1 | 2 | 3 | 4 | 5 | DEF]
River level BELOW 724 feet 6 inches

☐ **HA1.8** [1 | 2 | 3 | 4 | 5 | DEF]
Report to control room of VISIBLE DAMAGE affecting a Safe Shutdown/Vital Area

UNUSUAL EVENT

HU1 Natural and Destructive Phenomena Affecting the Protected Area

☐ **HU1.1** [1 | 2 | 3 | 4 | 5 | DEF]
Earthquake detected per AOP 901, Earthquake

☐ **HU1.2** [1 | 2 | 3 | 4 | 5 | DEF]
Report of a tornado touching down within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern

☐ **HU1.3** [1 | 2 | 3 | 4 | 5 | DEF]
Report of winds greater than 95 mph within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern

☐ **HU1.4** [1 | 2 | 3 | 4 | 5 | DEF]
Vehicle crash into plant structures or systems within the Plant Protected Area with NO confirmed damage to a Safe Shutdown/Vital Area or Control Room indication of degraded performance of a System of Concern

☐ **HU1.5** [1 | 2 | 3 | 4 | 5 | DEF]
Report of an unanticipated explosion within the Plant Protected Area resulting in visible damage to permanent structures or equipment

☐ **HU1.6** [1 | 2 | 3 | 4 | 5 | DEF]
Report of turbine failure resulting in casing penetration or damage to turbine or generator seals

☐ **HU1.7** [1 | 2 | 3 | 4 | 5 | DEF]
River level ABOVE 757 feet

☐ **HU1.8** [1 | 2 | 3 | 4 | 5 | DEF]
Uncontrolled flooding in a Safe Shutdown/Vital Area that has the potential to affect safety related equipment needed for the current operating mode

☐ **HU1.9** [1 | 2 | 3 | 4 | 5 | DEF]
River level BELOW 725 feet 6 inches

Fire or Explosion

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

HA2 Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

☐ **HA2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Fire or explosion in any Safe Shutdown/Vital Area
AND
Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area

UNUSUAL EVENT

HU2 Fire Within Protected Area Boundary Not Extinguished Within 15 Minutes of Detection

☐ **HU2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Fire in buildings or areas contiguous to any Safe Shutdown/Vital Area not extinguished within 15 minutes of control room notification or verification of a control room alarm

Toxic and Flammable Gas

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

HA3 Release of Toxic or Flammable Gases Within or Contiguous to a Vital Area Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown

☐ **HA3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Report or detection of toxic gases within or contiguous to a Safe Shutdown/Vital Area in concentrations that may result in an atmosphere Immediately Dangerous to Life and Health (IDLH)

☐ **HA3.2** [1 | 2 | 3 | 4 | 5 | DEF]
Report or detection of gases in concentration greater than the Lower Flammability Limit within or contiguous to a Safe Shutdown/Vital Area

UNUSUAL EVENT

HU3 Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant

☐ **HU3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect normal plant operations

☐ **HU3.2** [1 | 2 | 3 | 4 | 5 | DEF]
Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event

Security

GENERAL EMERGENCY

HG1 HOSTILE ACTION resulting in loss of physical control of the facility

☐ **HG1.1** [1 | 2 | 3 | 4 | 5 | DEF]
A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of ANY Safe Shutdown/Vital Area such that operation of equipment required for safe shutdown is lost.

☐ **HG1.2** [1 | 2 | 3 | 4 | 5 | DEF]
A HOSTILE ACTION has caused failure of Spent Fuel Pool Cooling systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in the pool.

SITE AREA EMERGENCY

HS4 HOSTILE ACTION within the PROTECTED AREA

☐ **HS4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by DAEC Security Shift Supervision.

ALERT

HA4 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

☐ **HA4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by DAEC Security Shift Supervision.

☐ **HA4.2** [1 | 2 | 3 | 4 | 5 | DEF]
A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

☐ **HA4.3** [1 | 2 | 3 | 4 | 5 | DEF]
DAEC Security Supervision reports any of the following security events that persist for 30 minutes, or greater, affecting the Plant Protected Area:
- Suspicious FIRE or EXPLOSION
- Significant Security System Hardware Failure
- Loss of Guard Post Contact

UNUSUAL EVENT

HU4 Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant.

☐ **HU4.1** [1 | 2 | 3 | 4 | 5 | DEF]
A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by DAEC Security Shift Supervision.

☐ **HU4.2** [1 | 2 | 3 | 4 | 5 | DEF]
A credible site specific security threat notification..

☐ **HU4.3** [1 | 2 | 3 | 4 | 5 | DEF]
A validated notification from NRC providing information of an aircraft threat.

Control Room Evacuation

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established

☐ **HS2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Control Room evacuation has been initiated
AND
Control of the plant cannot be established per AOP 915 within 20 minutes

ALERT

HA5 Control Room Evacuation Has Been Initiated

☐ **HA5.1** [1 | 2 | 3 | 4 | 5 | DEF]
Entry into AOP 915 for control room evacuation

UNUSUAL EVENT: None

Emergency Director Judgment

GENERAL EMERGENCY

HG2 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency

☐ **HG2.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

SITE AREA EMERGENCY

HS3 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency

☐ **HS3.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary

ALERT

HA6 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert

☐ **HA6.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels

UNUSUAL EVENT

HU5 Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE

☐ **HU5.1** [1 | 2 | 3 | 4 | 5 | DEF]
Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

Cold SD/ Refuel System Malfunct.

Loss of Power

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

CA3 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses

☐ **CA3.1** [4 | 5 | DEF]
Loss of power to or from the Startup or Standby Transformer resulting in a loss of all offsite power to Emergency Busses 1A3 and 1A4
AND
Failure of A Diesel Generator (1G-31) and B Diesel Generator (1G-21) to supply power to emergency busses 1A3 and 1A4
AND
Failure to restore power to at least one emergency bus, 1A3 or 1A4, within 15 minutes from the time of loss of both offsite and onsite AC power

UNUSUAL EVENT

CU3 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes

☐ **CU3.1** [4 | 5]
Loss of power to or from the Startup or Standby Transformer resulting in a loss of all offsite power to Emergency Busses 1A3 and 1A4 that is expected to last for greater than 15 minutes
AND
At least one Emergency Bus, 1A3 or 1A4, is powered by it's Standby Diesel Generator

CU7 Unplanned Loss of Required DC Power For Greater Than 15 Minutes

☐ **CU7.1** [4 | 5]
Unplanned Loss of Vital DC power to required DC busses based on bus voltage LESS THAN 105 VDC indicated
AND
Failure to restore power to at least one required DC bus within 15 minutes from the time of loss

RPV Level

GENERAL EMERGENCY

CG1 Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV

☐ **CG1.1** [4 | 5]
(1) Loss of RPV inventory as indicated by unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase
AND
(2) RPV Level:
a. LESS THAN +15 inches for GREATER THAN 30 minutes
OR
b. Cannot be monitored with Indication of core uncovery for GREATER THAN 30 minutes as evidenced by one or more of the following:
- Containment High Range Rad Monitor reading GREATER THAN 10Rem/hr
- Erratic Source Range Monitor Indication

AND
(3) Indication of Secondary Containment challenged as indicated by one or more of the following:
- Drywell Hydrogen or Torus Hydrogen GREATER THAN 6% AND Drywell Oxygen or Torus Oxygen GREATER THAN 5%
- Containment Pressure GREATER THAN 53 psig
- Secondary Containment not established
- Two or more Reactor Building areas exceed Max Safe Radiation Levels

SITE AREA EMERGENCY

CS1 Loss of RPV Inventory Affecting Core Decay Heat Removal Capability

☐ **CS1.1** [4]
With Secondary Containment not established:
a. RPV inventory as indicated by RPV level is LESS THAN 113.5 inches
OR
b. RPV level cannot be monitored for GREATER THAN 30 minutes with a loss of RPV inventory as indicated by unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase

☐ **CS1.2** [4]
With Secondary Containment established:
a. RPV inventory as indicated by RPV level is LESS THAN +15 inches
OR
b. RPV level cannot be monitored for GREATER THAN 30 minutes with a loss of RPV inventory as indicated by either:
- Unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase
- Erratic Source Range Monitor Indication

CS2 Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV

☐ **CS2.1** [5]
With SECONDARY CONTAINMENT NOT ESTABLISHED, EITHER of the following occurs:
a. RPV inventory as indicated by RPV level is LESS THAN 113.5 inches
OR
b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following:
- Containment High Range Rad Monitor reading GREATER THAN 10Rem/hr
- Erratic Source Range Monitor Indication

☐ **CS2.2** [5]
With SECONDARY CONTAINMENT ESTABLISHED, EITHER of the following occurs:
a. RPV inventory as indicated by RPV level is LESS THAN +15 inches
OR
b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following:
- Containment High Range Rad Monitor reading GREATER THAN 10Rem/hr.
- Erratic Source Range Monitor Indication

ALERT

CA1 Loss of RCS Inventory

☐ **CA1.1** [4]
Loss of RCS inventory as indicated by RPV level LESS THAN 119.5 inches

☐ **CA1.2** [4]
Loss of RCS inventory as indicated by unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase and RCS level cannot be monitored for GREATER THAN 15 minutes

CA2 Loss of RPV Inventory with Irradiated Fuel in the RPV

☐ **CA2.1** [5]
Loss of RPV inventory as indicated by RPV level LESS THAN 119.5 inches

☐ **CA2.2** [5]
Loss of RPV inventory as indicated by unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase and RPV level cannot be monitored for GREATER THAN 15 minutes

UNUSUAL EVENT

CU2 Unplanned Loss of RCS Inventory with Irradiated Fuel in the RPV

☐ **CU2.1** [5]
Unplanned RCS level decrease BELOW the RPV flange for 15 minutes or longer

☐ **CU2.2** [5]
RPV Level cannot be monitored
AND
Loss of RPV inventory as indicated by unexplained Drywell/Reactor Building Equipment or Floor Drain sump, or Torus, level increase

RCS Temp.

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

CA4 Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV

☐ **CA4.1** [4 | 5]
With Secondary Containment and RCS integrity not established, an unplanned event results in RCS temperature GREATER THAN 212 ºF

☐ **CA4.2** [4 | 5]
With Secondary Containment established and either RCS integrity not established or RCS inventory reduced, an unplanned event results in RCS temperature GREATER THAN 212 ºF for GREATER THAN 20 minutes. *(Note: If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.)*

☐ **CA4.3** [4 | 5]
An unplanned event results in RCS temperature GREATER THAN 212 ºF for GREATER THAN 60 minutes or results in an RCS pressure increase of GREATER THAN 10 psig. *(Note: If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.)*

UNUSUAL EVENT

CU4 Unplanned Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV

☐ **CU4.1** [4 | 5]
An unplanned event results in RCS temperature GREATER THAN 212 ºF

☐ **CU4.2** [4 | 5]
Loss of all RCS temperature and RPV level indication for GREATER THAN 15 minutes

Comm.

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

CU6 Unplanned Loss of All Onsite or Offsite Communications Capabilities

☐ **CU6.1** [4 | 5]
Loss of ALL of the following onsite communication capabilities affecting the ability to perform routine operation:
- Plant Operations Radio System
- In-Plant Telephones
- Plant Paging System

☐ **CU6.2** [4 | 5]
Loss of ALL of the following offsite communications capability:
- All telephone lines (commercial)
- Microwave Phone System
- FTS Phone System

Fuel Clad Degradation

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT: None

RCS Leakage

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

CU1 RCS Leakage

☐ **CU1.1** [4]
Unidentified or pressure boundary leakage GREATER THAN 10 gpm

☐ **CU1.2** [4]
Identified leakage GREATER THAN 25 gpm

Inadvertent Criticality

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

CU8 Inadvertent Criticality

☐ **CU8.1** [4 | 5]
An UNPLANNED extended positive period observed on nuclear instrumentation

ISFSI Events

Cask Confine. Boundary

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

EU1 Damage To A Loaded Cask Confinement Boundary

☐ **EU1.1**
Any one of the following natural phenomena events with resultant visible damage to or loss of a loaded cask confinement boundary:
- Report by plant personnel of a tornado strike
- Report by plant personnel of a seismic event

☐ **EU1.2**
The following accident condition with resultant visible damage to or loss of a loaded cask confinement boundary:
- A loaded transfer cask is dropped as a result of normal handling or transporting

☐ **EU1.3**
Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask confinement boundary

Security

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT: None

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. AND A.2 ----------NOTE--------- Only applicable for Function 7.a. -------------------------- Inhibit containment spray system.	12 hours for Functions 2.a, 2.b, 6.b, and 6.c AND 24 hours for Functions other than Functions 2.a, 2.b, and 6.b, and 6.c 24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL). <u>OR</u> D.2.1 Be in MODE 3. <u>AND</u> D.2.2 Be in MODE 4.	12 hours 12 hours 36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. [Deleted]		
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	I.1 Declare Standby Liquid Control (SLC) System inoperable. OR I.2 Isolate the Reactor Water Cleanup System.	1 hour 1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	J.1 Initiate action to restore channel to OPERABLE status.	Immediately
	<u>OR</u>	
	J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.	Immediately
K. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	K.1 -----------NOTE------------ Only applicable if inoperable channel is not in trip. ------------------------------	
	Declare associated Suppression Pool Cooling/Spray subsystem(s) inoperable.	Immediately
	<u>OR</u>	
	K.2 ----------NOTE------------ Only applicable if inoperable channel is in trip. ------------------------------	
	Declare Primary Containment inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	L.1 Isolate the primary containment vent and purge penetration flow paths.	1 hour
	OR	
	L.2 Establish administrative control of the primary containment vent and purge valves using continuous monitoring of alternate instrumentation.	1 hour

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Function 5.a; and (b) for up to 6 hours for Functions other than 5.a provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	Perform CHANNEL CHECK.	24 hours
SR 3.3.6.1.3	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.6.1.4	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.6	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.6.1.7	Perform CHANNEL CALIBRATION.	12 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.8	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.1.9	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 38.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≥ 821 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≤ 138% rated steam flow
d. Condenser Backpressure - High	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 7.2 inches Hg vacuum
e. Main Steam Line Tunnel Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F
f.. Turbine Building Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F

(continued)

(a) When any turbine stop valve is greater than 90% open or when the key-locked bypass switch is in the NORM Position.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\geq$ 165.6 inches
b. Drywell Pressure - High	1,2,3	2	H	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 2.2 psig
c. Offgas Vent Stack - High Radiation	1[(c)], 2[(c)], 3[(c)]	1	L	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	(b)
d. Reactor Building Exhaust Shaft – High Radiation	1,2,3	1	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 12.8 mR/hr
e. Refueling Floor Exhaust Duct – High Radiation	1,2,3	1	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 10.6 mR/hr
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 409 inches (inboard) $\leq$ 110 inches (outboard)

(continued)

(b) Allowable value is determined in accordance with the ODAM.

(c) During venting or purging of primary containment.

Table 3.3.6.1-1 (page 3 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
b. HPCI Steam Supply Line Pressure – Low	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 50 psig and ≤ 147.1 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 2.5 psig
d. Drywell Pressure - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
e. Suppression Pool Area Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 153.3°F
f. HPCI Leak Detection Time Delay	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	N/A
g. Suppression Pool Area Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F
h. HPCI Equipment Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 178.3°F
i. HPCI Room Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F

(continued)

Table 3.3.6.1-1 (page 4 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 164 inches (inboard) ≤ 159 inches (outboard)
b. RCIC Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 50.3 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.9	≥ 3.3 psig
d. Drywell Pressure – High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
e. RCIC Suppression Pool Area Ambient Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 153.3°F
f. RCIC Leak Detection Time Delay	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	N/A
g. RCIC Suppression Pool Area Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F
h. RCIC Equipment Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 178.3°F
i. RCIC Room Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F

(continued)

Table 3.3.6.1-1 (page 5 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 59 gpm
b. Area Temperature - High	1,2,3	1(d)	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 133.3°F
c. Area Ventilation Differential Temperature – High	1,2,3	1(d)	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	
RWCU Pump Room RWCU Pump A Room RWCU Pump B Room RWCU Heat Exch. Room					≤ 22.5°F ≤ 23.5°F ≤ 34.5°F ≤ 51.5°F
d. SLC System Initiation	1,2	1(e)	I	SR 3.3.6.1.9	NA
e. Reactor Vessel Water Level – Low Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≥ 112.65 inches
f. Area Near TIP Room Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 115.7°F
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≤ 152.7 psig
b. Reactor Vessel Water Level – Low	3,4,5	2(f)	J	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 165.6 inches
c. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
7. Containment Cooling System Isolation					
a. Containment Pressure – High	1,2,3	4	K	SR 3.3.6.1.3 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 1.25 psig

(d) Each Trip System must have either an OPERABLE Function 5.b or an OPERABLE Function 5.c channel in both the RWCU pump area and in the RWCU heat exchanger area.

(e) SLC System Initiation only inputs into one of the two trip systems.

(f) Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.