# SSES LOC 19 NRC Exam

1 Unit 1 startup was in progress with Reactor Power at 74% when RECIRC PUMP A DSCH HV-143-F031A open position logic failed causing a run back. The limit switch logic problem has been repaired and the runback logic is to be reset.

Which of the following actions must be completed to reset the runback logic?

A. lower GEN 1A & 1B SPEED control signal to slightly lower Recirc pump speed, reset limiter #~~2~~ 1, monitor for speed change.

B. lower GEN 1A SPEED control signal to slightly lower Recirc pump speed, reset limiter #2, monitor for speed change.

C. lower GEN 1A SPEED control signal to slightly lower Recirc pump speed, reset limiter #1 & #2, monitor for speed change.

D. lower GEN 1A SPEED control signal to slightly lower Recirc pump speed, reset limiter #1, monitor for speed change.

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**Question Data**

**Answer: D** lower GEN 1A SPEED control signal to slightly lower Recirc pump speed, reset limiter #1, monitor for speed change.

**Explanation/Justification:**

**A.** Runback does not affect both pumps, only 'A'
**B.** #2 limiter is caused by Low Reactor Level or Circ water pump trip
**C.** Valve position does not feed both limiters.
**D.** Correct answer

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295001** | **Partial or Complete Loss of Forced Core Flow Circulation** | **Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:** | **Recirculation flow control system** |

K/A# **295001.AA1.05** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-164-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

2 Unit 1 & 2 are operating at 100% power with a normal electrical lineup when a loss of Startup Bus 10 occurs.

Assuming no Operator actions, and all equipment functions as designed. What will be the status of the Unit 1 'A' ESS bus breakers following this event?

A. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Closed
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Closed

B. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

C. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Closed
D/G A to Bus 1A - (1A20104) Breaker Open

D. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Closed
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

## Question Data

**Answer: C** Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Closed
D/G A to Bus 1A - (1A20104) Breaker Open

**Explanation/Justification:**

**A.** No power available through 01 breaker and 04 breaker would not close if 01 breaker still closed on bus.
**B.** The D/G output breaker would be expected to be closed if both the 01 and 09 breakers open.
**C.** Correct answer
**D.** No power available through 01 breaker

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295003** | **Partial or Complete Loss of A.C. Power** | **Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following:** | **Emergency generators** |

test the fact that 201 bkr will close, but EDG breaker will should not have to close

K/A# **295003.AK2.02** K/A Importance **4.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-003-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

(Chord stem to distinguish power supply is lost before initiation signal (logic not accomplished prior to loss of power))

3 Which of the following describes the effect of losing 125 VDC 1D614 power to the Auto Depressurization System (ADS) system with a valid initiation signal present?

A. No ADS valves will open

B. ADS logic "B" will still initiate ADS, Only 3 ADS valves will open

C. ADS logic "A" will receive backup power from 125 VDC 1D6~~3~~4, All ADS valves will open

D. ADS logic "B" will still initiate ADS, All ADS valves will open

## Question Data

**Answer: D** ADS logic "B" will still initiate ADS, All ADS valves will open

**Explanation/Justification:**

**A.** All ADS valves will open.
**B.** 6 ADS valves will open
**C.** 1D614 does not have an alternate
**D.** correct answer, Either division of ADS will provide actuation of all 6 ADS valves

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295004** | **Partial or Complete Loss of D.C. Power** | **Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:** | **Electrical bus divisional separation** |

K/A# **295004.AK1.03** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-083**

Question Source: **Modified** **Fitzpatrick, 1992** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

4 The following alarm is received for Division II;

TURB STOP VLV CLOSURE TRIP AR-104-E02

Which of the following lists the items causing the alarm?

typo

A. Turbine tripped
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

B. Turbine tripped
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel A.

C. Turbine reset and speed not selected.
Stop Valves or Control Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

D. Turbine reset and speed not selected.
Stop Valves 1 and 3 or 2 and 4 < 98% open.
Loss of power to RPS Channel B.

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**Question Data**

**Answer: A** Turbine tripped
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

**Explanation/Justification:**

**A.** Correct answer
**B.** Loss of power from B RPS causes Div 2 alarm.
**C.** Control valves will not cause alarm.
**D.** Valve position is <95%

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295005** | **Main Turbine Generator Trip** | **Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:** | **Turbine valve position** |

K/A# **295005.AA2.03** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-104-001**

Question Source: **New** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

5 During a normal plant shutdown, a reactor scram is manually inserted, which of the following criteria is utilized to determine if EO-100-113, "Level/Power Control" entry is also required? | typo

A. The status of the Average Power Range Monitor (APRM) "Downscale" lights.

B. The value of reactor Source Range Monitor (SRM) period after rod movement and detector insertion is complete.

C. The position and number of control rods inserted.

D. The ability to monitor instrumentation for valid, current reactor power level.

---

**Question Data**

**Answer: C** The position and number of control rods inserted.

**Explanation/Justification:**

**A.** APRM downscales not in is EO-102 entry (>5% power)

**B.** SRM period provides indication that the reactor is "shutting down" not "shutdown"

**C.** correct answer, >1 rod >00, enter EO-113, no EO-102 entry required

**D.** power not known is EO-102 entry

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295006** | **SCRAM** | **Knowledge of the operational implications of the following concepts as they apply to SCRAM:** | **Shutdown margin** |

K/A# **295006.AK1.02** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-100-101**

Question Source: **Modified** **Susquehanna, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

references will be modified EOPs:
no EOP entry conditions will be given w/ references

but EO 100-102 gives kick-out to -113 if all rods don't go in

∴ too easy

will be replaced

# SSES LOC 19 NRC Exam

6 ON-100-009, PLANT SHUTDOWN FROM OUTSIDE CONTROL ROOM, directs opening HPCI TEST LINE ISO VLV HV-155-F011.

The basis for opening HV-155-F011 is to:

A. allow RCIC System to be cycled between level control and pressure control.

B. provide a flow path for HPCI in the full flow test mode if it is needed for reactor pressure control.

C. ensure that keepfill protection is available for both HPCI and RCIC.

D. permit use of HPCI if RCIC is unable to provide sufficient makeup.

---

**Question Data**

**Answer: A** allow RCIC System to be cycled between level control and pressure control.

**Explanation/Justification:**

**A.** correct answer,
**B.** Valve position does not affect HPCI Flow path.
**C.** Does not affect keep fill for HPCI
**D.** HPCI runs in auto from -38 to +54.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295016** | **Control Room Abandonment** | **Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT:** | **Disabling control room controls** |

K/A# **295016.AK3.03** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-100-009**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

KA not valid e Surg.
quest. will be replaced
after new KA 295016 is randomly selected

# SSES LOC 19 NRC Exam

7 Given the following conditions:

- Unit 2 is operating at 100% power
- The "2B" Reactor Protection System (RPS) Bus is on the alternate power supply
- The "2A" RPS MG Set has just tripped

Unit 2 may operate in Mode 1 for a limited amount of time based upon:

A. the length of time for restoration of Emergency Switchgear cooling.

B. the availability of the Reactor Building Sump Pumps.

C. the Containment instrument gas supply to the Inboard MSIVs.

D. the availability of the Reactor Recirculation Pumps.

lose RPS → lose CIU for C16 → comp trip → lose supply to 1B MSIV's

**Question Data**

**Answer: D** the availability of the Reactor Recirculation Pumps.

**Explanation/Justification:**

**A.** cooling will be restored if 'A' equipment in service or no effect if 'B' equipment in service.
**B.** continued operation based on being able to complete leakage surveillance, not an immediate concern
**C.** not an immediate concern
**D.** correct answer, loss of RPS causes containment isolation and loss of cooling water to the recirc pumps

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295018.2.4.11** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-158-001**

Question Source: **Modified** **SSES, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

2.4.11 is generic for K/A 295018 "Partial or total loss of CCW" - see ref on next page ∴ ok

# SSES LOC 19 NRC Exam

8 With Unit 1 at full power, annunciator AR-124-B01, INSTRUMENT AIR HEADER LO PRESSURE alarms.

What is this alarm actually monitoring and what is expected to happen if the condition continues to degrade?

A. Scram air header pressure is less than 65 psig. If not corrected, Scram Discharge Volume Vent and Drain Valves will fail closed preventing a scram.

B. Scram air header pressure is less than 75 psig. If not corrected, the scram valves will begin to open, scraming in rods.

C. Instrument air header pressure is less than 80 psig. If not corrected, the scram valves will begin to open, drifting in rods.

D. Instrument air header pressure is less than 80 psig. If not corrected, Scram Discharge Volume Vent and Drain Valves will fail closed preventing a scram.

**Question Data**

**Answer: C** Instrument air header pressure is less than 80 psig. If not corrected, the scram valves will begin to open, drifting in rods.

**Explanation/Justification:**

**A.** Alarm is measuring IA header pressure SDV vent & drain valves closing will not prevent a scram.
**B.** Alarm is measuring instrument air header pressure.
**C.** correct answer ON-118-001 discussion section :scram inlet and outlet valves will begin drifiting open causing random control rod insertion.
**D.** SDV vent & drain valves closing will not prevent a scram.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295019.2.4.46** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-118-001** ✓

Question Source: **Modified** **Quad-Cities 1 & 2, 1996** Level Of Difficulty: (1-5) ~~3~~ 2 ✓

Question Cognitive Level: **Fundamental** ✓ 10 CFR Part 55 Content: **55.41**

Inst. air feeds scram air header

K/A generic ability to verify alarms consistent w/ plant conditions re I.A.

# SSES LOC 19 NRC Exam

9 Why is the operator directed to ensure Reactor water level greater than +45 inches when RHR Shutdown Cooling is lost?

A. to ensure natural circulation occurs since forced circulation is lost.

B. to ensure the feedwater spargers effectively mix the feedwater added to the Reactor Vessel.

C. to maintain Reactor Narrow range water level instrumentation on scale when the Reactor water temperature increases.

D. to ensure the water from the annulus area of the Reactor flows into the core shroud area as level decreases inside the shroud.

---

**Question Data**

**Answer: A** *to ensure natural circulation occurs since forced circulation is lost.*

**Explanation/Justification:**

**A.** correct answer, ON-149-001 discussion, >+45 raises water level above steam seperator to establish natural circulation

**B.** Any feedwater addition has no affect on why level is maintained >+45 – needed for ATWS, not SDC.

**C.** Narrow and wide range upscale <200 psig in vessel with actual level >+45

**D.** with level >+45 there is no difference in water level inside or outside of shroud.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295021** | **Loss of Shutdown Cooling** | **Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING:** | **Establishing alternate heat removal flow paths** |

K/A# **295021.AK3.05** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-149-001**

Question Source: **Modified** **Monticello 1, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

10 Given the following conditions:

- A Railroad Access Shaft Exhaust Duct Radiation High signal is received.
- Standby Gas Treatment responds as required.
- All Zone I differential pressures are -.27" WG.

Select the Standby Gas Treatment system response to Zone II differential pressure decreasing to -.23" WG.

A. Outside Cooling air dampers will modulate open.

B. Outside makeup air dampers will modulate closed.

C. Standby Gas Treatment dampers will not adjust for Zone II delta P.

D. Standby gas treatment fan inlet vanes will modulate open.

---

**Question Data**

**Answer: C** Standby Gas Treatment dampers will not adjust for Zone II delta P.

**Explanation/Justification:**

**A.** Outside air damplers modulate based on gas stream temperature., not any Z-I, II or III delta pressure.
**B.** Outside air damplers modulate based on suction pressure, not Z-I, II or III delta pressure.
**C.** correct answer, relay circuit excludes the signal from a zone that does not have an isolation signal present
**D.** Inlet vanes vary position to maintain flow rate

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295023** | **Refueling Accidents** | **Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:** | **Standby gas treatment/FRVS** |

K/A# **295023.AA1.07** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-070**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

11 Given the following parameters:

| | |
|---|---|
| - Drywell pressure | 3.5 psig and rising |
| - Drywell temperature | 145 degrees F and rising |
| - Suppression chamber pressure | 4.6 psig and rising |
| - Suppression pool water temperature | 87 degrees F and steady |

What event from below would cause the conditions listed above?

A. A safety relief valve tail pipe has broken in the Suppression Chamber while the valve is open.

B. A pipe break into the drywell has occurred with a suppression chamber to drywell vacuum breaker open.

C. A downcomer vacuum breaker has failed open during a recirculation leak to the drywell.

D. A recirculation line partial break has occurred with all containment parameters responding as designed.

---

**Question Data**

**Answer: A** A safety relief valve tail pipe has broken in the Suppression Chamber while the valve is open.

**Explanation/Justification:**

**A.** correct answer, energy into chamber but not into pool, vacuum breakers opening back to drywell when d/p high enough

**B.** downcomer vacuum breakers are designed to be open for these conditions, equalize pressure across the drywell floor when drywell pressure less than chamber pressure

**C.** only one vacuum breaker failing would not provide a vent path to Suppression pool atmosphere.

**D.** all parameters way too low, especially pool temperature

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295024** | **High Drywell Pressure** | **Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:** | **Suppression chamber pressure: Plant-Specific** |

K/A# **295024.EA2.04** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References:

Question Source: **Modified** **Susquehanna, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

12 The reason for maintaining suppression pool temperature below the Heat Capacity Temperature Limit EOP-100-103, PRIMARY CONTAINMENT CONTROL is:

A. to assure primary containment vent valve opening capability following RPV depressurization

"A" may be a subset of "D"

B. to provide adequate subcooling in the pool to prevent chugging of the SRV downcomers.

C. to assure the containment design pressure will not be exceeded due to compression of the non-condensable gasses.

D. to assure the containment design pressure will not be exceeded following RPV depressurization

Close may partially correct

**Question Data**

**Answer: A** to assure primary containment vent valve opening capability following RPV depressurization

**Explanation/Justification:**

**A.** correct answer
**B.** suppression chamber pressure of 13 psig pertains to chugging of SRVs
**C.** assumption that all non-condensables are in the suppression chamber, reason for shape of curve, not reason for curve
**D.** the concern is vent valve operation not design pressure of the containment

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295025.2.4.18** K/A Importance **2.7** ~~Exam~~ Level **RO**
References provided to Candidate **None** Technical References: **EO-000-103**
Question Source: **Modified** **Clinton 1, 2000** Level Of Difficulty: (1-5) **3**
Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

K/A = generic, will be supplied.

Will consider new D. distractor

# SSES LOC 19 NRC Exam

13 Given the following conditions:

- Unit 1 is in an ATWS condition
- Reactor pressure is 950 psig
- Suppression Pool temperature is 175 degrees F

Based on the ~~current reactor pressure and Suppression Pool temperature, if Suppression Pool water level began to decrease~~ given conditions, when would an emergency blowdown be REQUIRED after the rods are inserted?

A. 19.5 feet

B. 22 feet

C. 17 feet

D. 15.5 feet

**Question Data**

**Answer: D** 15.5 feet

**Explanation/Justification:**

**A.** 19.5 is RCIC room equalization value.
**B.** minimum normal level of the suppression pool.
**C.** at 17 feet go to RPV control
**D.** *Correct answer, using HCTL curve.*

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295026** | **Suppression Pool High Water Temperature** | ***Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:*** | **Suppression pool level** |

K/A# **295026.EA2.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-000-103**

Question Source: **Modified** **Peach Bottom, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

EOP uses HCTL curve to "interpret" SP level needed for blowdown based on SP temp.

# SSES LOC 19 NRC Exam

14 What would be the consequence of NOT limiting Drywell Spray flow for the first 30 seconds of operation?

NOT limiting Drywell Spray flow for the first 30 seconds of operation will:

A. cause the Drywell Spray Initiation Limit (DWSIL) of 13 psig to be exceeded.

B. cause a pressure drop fast enough to exceed the Containment design differential pressure.

C. cause rapid pressure drop with minimal Suppression Chamber vapor to support Sup Pool - Drywell vacuum breaker operation.

D. cause excessive thermal and mechanical shock to the Drywell downcomers.

---

**Question Data**

**Answer: B** cause a pressure drop fast enough to exceed the Containment design differential pressure.

**Explanation/Justification:**

**A.** limiting flow allows initiation of sprays at any temperature (not pressure), or without concern in all regions of DWSIL curve.
**B.** Correct answer.
**C.** Vapor in drywell is issue not suppression pool
**D.** not a consideration

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295028** | **High Drywell Temperature** | **Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE:** | **Drywell spray: Mark-I&II** |

K/A# **295028.EA1.01** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-103**

Question Source: **Modified** **OYSTER CREEK, 1991** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

15 The plant is in a LOCA with ECCS systems injecting to the reactor.

Suppression Pool level has dropped to 20.5 feet.

Which of the following is a condition that exists due to Suppression Pool level?

A. Suppression Pool average temperature ~~cannot be determ~~ is not indicated. ined.

B. The SRV tailpipe T-Quenchers have been uncovered.

C. The HPCI Turbine Exhaust has been uncovered.

D. Containment Pressure ~~cannot be determ~~ is not indicated ined.

---

**Question Data**

**Answer: A** *Suppression Pool average temperature cannot be determined.*

**Explanation/Justification:**

**A.** *Correct answer, must use lower SPOTMOS sensors.*
**B.** Not applicable, SRV tailpipe ehausts uncover at 5'
**C.** HPCI Turbine exhaust is a concern at <17'
**D.** Contsainment pressure idication is available.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295030** | **Low Suppression Pool Water Level** | **Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:** | **Suppression pool temperature** |

K/A# **295030.EA2.02** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-103**

Question Source: **Modified** **Grand Gulf 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

K/A temp not available (determinable or interpretable) due to low SP water level

# SSES LOC 19 NRC Exam

16 Under certain ATWS conditions, the EOPs direct the operators to take action to deliberately lower RPV level in order to reduce reactor power.

Which of the following describes why reactor power decreases as RPV level is lowered?

Lowering RPV level

A. reduces the natural circulation flow which increases the rate of steam removal, resulting in an increase in Void fraction which adds negative reactivity.

B. uncovers the feedwater spargers to reduce core inlet subcooling, resulting in an increase in Void fraction which adds negative reactivity.

C. reduces the amount of carryover through the steam dryer and separators, resulting in a higher Void fraction, adding negative reactivity. plausible?

D. reduces the NPSH available to the Jet pumps, reducing flow the pumps deliver, resulting in a ~~lower quality~~ higher Void fraction, adding negative reactivity.

**Question Data**

**Answer: B** uncovers the feedwater spargers to reduce core inlet subcooling, resulting in an increase in Void fraction which adds negative reactivity.

**Explanation/Justification:**

**A.** steaming rate decreases due to increased core inlet temperature and reduced moderation
**B.** Correct answer
**C.** steaming rate decreases as power decreases, total carryover decreases
**D.** quality is a measure of steam to moisture, not applicable to core inlet parameters

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 295031 | **Reactor Low Water Level** | **Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL:** | **Natural circulation: Plant-Specific** |

K/A# **295031.EK1.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-100-113**

Question Source: **Modified** **WPPSS 2, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

will look e C. + D.

# SSES LOC 19 NRC Exam

new distractors
make sure challenges
steam cooling possibility
(steam twist)

17 Following a LOCA on Unit 1 the following conditions exist:

- All control rods have inserted.
- RPV water level is -162 inches (actual level) and slowly decreasing
- RPV pressure is 815 psig and stable
- There are NO injection sources available

Which of the following will maintain STEAM COOLING?

A. The reactor has been rapidly depressurized per EO-100-112. Steam Sys 815 psig

B. RPV water level remains between -161 and -205 inches.

C. RPV pressure remains between 815 psig and 1087 psig.

D. A Table 5 high-pressure source is lined up, started, and is injecting.. not plausible
No injection in Steam cooling

**Question Data**

**Answer: B** RPV water level remains between -161 and -205 inches.

**Explanation/Justification:**

**A.** Rapid depressurization accelerates the rate of inventory loss and pressure drop, thereby shortening the time that steam cooling is maintained.

**B.** correct answer, Steam cooling occurs when water heated in the core boils turns to steam and rises in the bundles cooling the upper portions. This occurs below -161 inches and continues to -205 inches (MZIRWL). For this to occur there must be a steam flow path and zero injection. From the stated conditions the break is providing a flow path (pressure is NOT rising).

**C.** RPV pressure must be stable or decreasing. Raising reactor pressure will be detrimental to the steam cooling. If RPV pressure is rising, the assumptions of the MZIWL calculation are no longer valid and the core may not be adequately cooled.

**D.** If an injection source is line up and running (injecting), the assumptions of the MZIWL calculation are no longer valid and the core may not be adequately cooled. RPV depressurization is required to maximize injection flow rate and cool the core by submergence.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295031** | **Reactor Low Water Level** | **Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:** | **Adequate core cooling** |

K/A# **295031.EA2.04** K/A Importance **4.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-102**

Question Source: **Modified** **Susquehanna, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

Mem

# SSES LOC 19 NRC Exam

Reach ? for KA

18 A failure to completely scram occurred on Unit 1 with Reactor Power at 15%.

- No high pressure systems are available.
- Reactor Water level is at -100" and slowly dropping with several rods not fully inserted into the core.

Why is Core Spray NOT a system listed in Table 15 of EO-100-113, LEVEL/POWER CONTROL for use to control water level during a a failure to scram transient?

A. cannot be throttled, not easy to control — Is this true Can't be throttled — plausible ?

B. injects inside the shroud

C. provides indirect core cooling

D. required time to line-up flowpath. NOT plausible

**Question Data**

**Answer: B** injects inside the shroud

**Explanation/Justification:**

**A.** cold water injects inside shroud, cause power changes.
**B.** Correct answer,
**C.** not during an ATWS situation
**D.** cold water on fuel largest concern.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 295037 | **SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown** | **Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:** | **Reactor water level** ? |

not a good match this has mod for dir with ~~Sat~~ injects during ATWS to control power

K/A# **295037.EK2.09** K/A Importance **4.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-100-113**

Question Source: **Modified** **Lasalle 1, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

EO given as ref in question #13

# SSES LOC 19 NRC Exam

19 EO-100-105, "Radioactivity Release Control", directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building except those systems required to support EOP/DSP actions.

These systems are specifically exempted from isolation because:

New disk. needed

A. these additional isolations would require an unnecessarily escalation of the emergency classification. no plausible

B. they are required to support alternate reactor depressurization methods. NOT plausible

C. additional ~~off-site~~ rad. consequences releases from them are unlikely. probably a true statement on face value

D. isolation may ultimately result in a much larger uncontrolled radiological release.

---

**Question Data**

**Answer: D** isolation may ultimately result in a much larger uncontrolled radiological release.

**Explanation/Justification:**

**A.** not a consideration for these conditions
**B.** not true
**C.** alternate depress methods are part of EOPs
**D.** correct answer

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295038** | **High Off-Site Release Rate** | **Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE:** | **System isolations** |

K/A# **295038.EK3.02** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-00-105**

Question Source: **Modified** **Susquehanna, 1999** Level Of Difficulty: (1-5) ~~1~~ 2

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

LOD 1-2, doesn't discriminate

# SSES LOC 19 NRC Exam

20 The T-20 Transformer developed an internal fault causing a fire and automatic actuation of the fire protection deluge system. The Fire Protection System functions as designed. For these conditions the Diesel Driven Fire pump will automatically start if the fire protection header pressure drops to:

Too many questions of this type
Setpoint questions
does not discriminate
LOD 1-2

A. 105 psig

B. 95 psig

C. 85 psig.

D. 125 psig

---

**Question Data**

**Answer: C** 85 psig.

**Explanation/Justification:**

**A.** The auto start of the Jockey Fire Pump
**B.** The auto start of the Motor Driven Fire pump
**C.** correct answer,
**D.** The auto shutdown of the Jockey Fire Pump

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **600000.2.1.31** K/A Importance **4.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-013**

Question Source: **New** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

K/A= generic - will be supplied

# SSES LOC 19 NRC Exam

21 Unit 2 is starting up with power at 50%.

The third Reactor Feedpump has just been placed in service.

A Reactor Recirc system runback to 30% has occurred. The runback is due to the failure of the Feedwater Level Control System total feed flow signal.

What is the expected plant response with no operator actions?

A. Reactor Recirc Pump Trip signal

B. High Water RPV level alarm

C. Feedwater level control system Setpoint Setdown signal

D. Low Reactor Water Level Alarm

---

**Question Data**

**Answer: B** High Water RPV level alarm

**Explanation/Justification:**

**A.** No trip signal to recirc pumps on high level.

**B.** Correct answer, Recirc runback caused by failure of feedwater flow signal to less than 20% flow, FWLC system sees a steam feed mismatch with feed flow low as compared to steam flow, FWLC will increase feed flow to cancel Steam/Feed mismatch UNTIL level signal error over-rides steam feed mismatch thus, vessel level will increase.

**C.** Level will fail high not low.

**D.** Level will fail high not low.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295008** | **High Reactor Water Level** | **Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL:** | **Feed flow/steam flow mismatch** |

K/A# **295008.AK1.03** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-045**

Question Source: **Modified** **Lasalle 1, 1996** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

22 Unit 1 is at 100% power in a normal lineup when annunciator RX WATER HI-LO LEVEL AR-101-B17 actuates.

The following indications are observed at 1C651:

- The Master Feed Water Level Controller LIC-C32-1R600 is in AUTO and set at 34.
- 1'A' Reactor Feed Pump speed controller SIC-C32-1R601A demand signal is slowly lowering.
- 1'B' Reactor Feed Pump speed controller SIC-C32-1R601B demand signal is slowly rising.
- 1'C' Reactor Feed Pump speed controller SIC-C32-1R601C demand signal is slowly rising.
- The failed RFP A speed controller is placed in MANUAL.

Which of the following is the CORRECT, operator response for this situation?

A. adjust feed flow to equal B & C Reactor Feedpumps, or, lower Motor Speed Changer, activate Hydraulic Jack, control speed to control flow.

B. adjust feed flow to equal B & C Reactor Feedpumps, or, activate Hydraulic Jack, lower Motor Speed Changer, control speed to control flow.

C. adjust Master Feed Water Level Controller to maintain RPV water level ~35", dedicate Operator for RPV level.

D. adjust feed flow to equal B & C Reactor Feedpumps, or, activate Hydraulic Jack, lower EAP, control speed to control flow.

good out

**Question Data**

**Answer: A** adjust feed flow to equal B & C Reactor Feedpumps, or, lower Motor Speed Changer, activate Hydraulic Jack, control speed to control flow.

**Explanation/Justification:**

**A.** correct answer,
**B.** Lower MSC first then set Hydraulic Jack.
**C.** 'A' RFP controller has failed, attempting to control level with master controller will not work.
**D.** Lower MSC first then set Hydraulic Jack.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295009** | **Low Reactor Water Level** | **Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL:** | **Reactor water level control** |

K/A# **295009.AA1.02** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-145-001**

Question Source: **Modified** **Duane Arnold 1, 1999** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

23 A Reactor scram resulted in water level dropping to -31 inches on Wide Range Level.

- Reactor level has since recovered to + 10 inches on Narrow Range.
- Reactor Pressure is being maintained with the Turbine Bypass Valves.
- The maximum Reactor Pressure during the transient was 1080 psig.
- The operators have placed the 'A' RHR pump in Suppression Pool Cooling.
- Suppression Pool Water Temperature is rising.

Which of the following is the cause of rising Suppression Pool water temperature?

A. RCIC in operation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET closed.

B. RCIC and HPCI in operation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET closed

C. HPCI in operation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET open

D. SRV actuation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET closed

**Question Data**

**Answer: A** RCIC in operation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET closed.

**Explanation/Justification:**

**A.** correct answer, some cooling being provided with some flow forced through heat exchanger, heat input from RCIC auto start at -30", level not low enough to auto start HPCI, pressure not high enough to open SRV.

**B.** HPCI did not receive a auto start signal.

**C.** Pressure did not get high enough to open SRV.

**D.** Pressure did not get high enough to open SRV.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295013** | **High Suppression Pool Temperature** | **Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:** | **Suppression pool cooling** |

K/A# **295013.AA1.01** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-149-005**

Question Source: **Modified** **Grand Gulf 1, 2000** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

24 Given the following on Unit 1:

- A reactor scram signal has occurred
- The full core display has several full-in lights NOT displayed
- The Four Rod Display is unavailable
- Computer rod position data is not available

You are directed to verify the scram and you are aware of industry experiences where several light bulbs on the full core display have failed concurrently.

Which of the following indications would cause you to conclude that the scram may NOT have been successful?

A. All LPRM "downscale" lights are illuminated

B. Reactor period stable at -80 seconds on SRMs

C. All APRM "downscale" lights are not illuminated

D. Div I and Div II RPS trip annunciators are lit on 1C601.

**Question Data**

**Answer: C** All APRM "downscale" lights are not illuminated

**Explanation/Justification:**

**A.** Expected downscale indication on scram
**B.** Expected negative period on scram
**C.** Correct answer
**D.** Expected indication on a scram

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295015** | **Incomplete SCRAM** | **Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM:** | **Control rod position** |

K/A doesn't really match question determined by S/D & APRM

K/A# **295015.AA2.02** K/A Importance **4.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-100-101**

Question Source: **Modified** **Hope Creek Unit 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

re-work, potentially pick new K/A (this K/A too easy to make good quest. from)

# SSES LOC 19 NRC Exam

25 The following statement exists in EO-100-105, RADIOACTIVITY RELEASE CONTROL:

" IF TURBINE OR RW BUILDING HVAC NOT IN SERVICE RESTART APPLICABLE HVAC AS REQ'D"

Which of the following is the basis for keeping the Turbine Building Ventilation System in operation while executing EO-100-105, RADIOACTIVITY RELEASE CONTROL?

Having Turbine Building Ventilation in service:

A. maintains Turbine Building pressure above Reactor Building Pressure.

B. prevents having an unmonitored ground release from the Turbine Building.

C. prevents a reactor scram due to high temperature in the MSL tunnel.

D. ensures adequate dilution of the gases discharged through the Turbine Building Vent.

---

**Question Data**

**Answer: B** prevents having an unmonitored ground release from the Turbine Building.

**Explanation/Justification:**

**A.** Turbine Building pressure has no corelation to Reactor Building pressure
**B.** correct answer, operation with no ventilation in service will lead to an unmonitored ground level release
**C.** the EOP is not addressing high temperature in the steam tunnel.
**D.** the EOP is not addressing dilution of releases from the ventilation system

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295017** | **High Off-Site Release Rate** | **Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE:** | **Protection of the general public** |

K/A# **295017.AK1.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-105**

Question Source: **Modified** **Nine Mile Point 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

26 Suppression Pool Cleanup is in operation discharging to Rad Waste to lower the level of the Unit 1 Suppression Pool.

An inadvertant trip of the Division I logic for High Drywell Pressure is actuated and confirmed by alarms received in the Control Room.

What would be the trend of the Rad Waste Collection Tank level indicator as a result of the High Drywell Pressure?

A. LRW Collection Tank Level trend is rising, Suppression Pool Level is lowering, Suppression Pool Cleanup pump tripped and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 and OB ISO HV-15768 are open.

B. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup pump tripped and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 is closed.

C. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup pump running and SUPP POOL WTR FILT PP SUCT OB ISO HV-15768 is open.

D. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup pump Running and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 and OB ISO HV-15768 are closed.

**Question Data**

**Answer: B** LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup pump tripped and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 is closed.

**Explanation/Justification:**

**A.** Suction valve would close stopping transfer to LRW
**B.** *correct answer, Pump trips and IB suction closes on Div I LOCA signal, stopping transfer to LRW.*
**C.** Pump would trip and LRW tank level constant
**D.** Pump trips, on Div I only IB valve closes.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295020** | **Inadvertent Containment Isolation** | **Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION:** | **Suppression pool water level response** |

K/A# **295020.AK3.06** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-159-002**

Question Source: **Modified** **Limerick 1, 1995** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

27 Unit 2 is operating at 90% power.

- The 2B CRD pump in service and the in-service Flow Control Valve has failed closed.

Which of the following are the expected alarms for these conditions?

A. AR-204-H05 ROD DRIFT
AR-207-C01 CRD PUMP SUCTION FILTER HIGH DIFF PRESS

B. AR-202-F03 RECIRC PUMP A SEAL CLG WATER LO FLOW
AR-202-F06 RECIRC PUMP B SEAL CLG WATER LO FLOW

C. AR-203-H05 CRD PANEL 1C007 HI TEMP
AR-207-A01, CRD CHARGING WATER HI PRESS

D. AR-207-C02 CRD PUMPS DRIVE WATER FLTR HI DIFF PRESS
AR-207-E02 CRD PUMP B MOTOR OVERLOAD

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**Question Data**

**Answer: C** AR-203-H05 CRD PANEL 1C007 HI TEMP
AR-207-A01, CRD CHARGING WATER HI PRESS

**Explanation/Justification:**

**A.** Flow control valve could cause rod drift if failed open, suction DP due tio high flow or strainer clogged.
**B.** Seal cooling alarm is RBCCW flow.
**C.** Correct answer, Flow control valve or Drive Pressure Control valve closed will cause alarm and cause decreased cooling water flow.
**D.** Hi DP caused by high flow or filter clogged, CRD pump will have reduced load and not trip.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295022** | **Loss of CRD Pumps** | **Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:** | **CRD mechanism temperatures** |

K/A# **295022.AA2.03** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-207-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

Network stem to check need for RHR injection

28 Unit 1 is operating at 75% power
-Valve stroke time testing is in progress on the "A" RHR Pump Suppression Pool Suction Valve HV151F004A
-HV151F004A is currently closed
-All other RHR system components are in their normal standby lineup
-A steam break causes drywell pressure to reach 2.0 psig

Which of the following describes the response of the F004A valve and the "A" RHR pump?

A. The HV151F004A valve automatically opens and the "A" RHR pump will run after the HV151F004A is fully open.

B. The HV151F004A valve must be opened and the "A" RHR pump must be manually started after HV151F004A is fully open.

C. The HV151F004A valve automatically opens but the "A" RHR pump must be started by the operator after HV151F004A is fully open.

D. The HV151F004A valve must be opened and the ,"A" RHR pump will auto start after HV151F004A is fully open.

---

**Question Data**

**Answer: B** The HV151F004A valve must be opened and the "A" RHR pump must be manually started after HV151F004A is fully open.

**Explanation/Justification:**

**A.** No auto action associated with the 04 valve.

**B.** correct answer, valve must be manually opened and pump may be started manually. Pump will not auto start since there is no low level or low reactor pressure

**C.** No auto action associated with the 04 valve.

**D.** No auto action associated with the 04 valve, Punp WOULD auto start if there were a low level or low pressure but no such signal is given in the initial conditions. Suction valve will not auto open.

2# DW NOT in Mode.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **203000** | **RHR/LPCI: Injection Mode (Plant Specific)** | **Knowledge of RHR/LPCI: INJECTION MODE design feature(s) and/or interlocks which provide for the following:** | **Adequate pump net positive suction head (interlock suction valve open): Plant-Specific** |

K/A# **203000.K4.06** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **Modified** **Hope Creek Unit 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

29 The following conditions exist during refueling operations:

- The Reactor Mode Switch is in REFUEL
- Irradiated fuel is in the reactor
- The reactor vessel head has been removed
- The Fuel Pool gates are installed
- Reactor water level is 18" above the reactor vessel flange

Which of the following defines the operability status required for shutdown cooling?

A. One shutdown cooling subsystem is required to be operable and in service. The other subsystem is NOT required to be operable.

B. Shutdown cooling is NOT required to be operable with RWCU in service.

C. Two subsystems of shutdown cooling are required to be operable and one subsystem is required to be in service.

D. Two subsystems of shutdown cooling are required to be operable and in service.

**Question Data**

**Answer: C** Two subsystems of shutdown cooling are required to be operable and one subsystem is required to be in service.

**Explanation/Justification:**

**A.** both shutdown cooling subsystems are required to be operable
**B.** shutdown cooling is required to be operable
**C.** correct
**D.** one subsystem is required to be in operation

don't like just section of TS. for RO's to determine. Consult.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| K/A# | 203000.2.2.22 | K/A Importance 3.4 | Exam Level RO |
| References provided to Candidate | TS 3.4.8 & 3.4.9 | Technical References: | SSES Unit 1 Tech Specs ~~3.9.8~~ 3.4.9 typo |
| Question Source: | Modified Limerick 1, 1996 | Level Of Difficulty: (1-5) | 3 |
| Question Cognitive Level: | Analysis | 10 CFR Part 55 Content: | 55.41 |

PICT

all Sect 3 without basis for RO's

K/A = generic, will be applied

all section 3 TS will be provided (check other questions for other answers in exam)

No work TS references LOD = 1 with

# SSES LOC 19 NRC Exam

30 Unit 1 is in cold shutdown with the "A" RHR pump in the shutdown cooling mode of operation. Reactor water level decreases to the LPCI system initiation setpoint.

What is the expected plant response assuming no operator action five minutes after the LPCI system initiation signal was generated ?

A. RHR loop suction and discharge valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A" & "C" RHR pump breakers are ~~pumping~~ (cycling open/closed).
The "B" and "D" RHR pumps are running but not injecting into the reactor vessel.

B. RHR loop suction and discharge valves, HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A", "B", "C" and "D" RHR pumps are injecting into the reactor vessel.

C. RHR loop suction and discharge valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO remain open.
The "A" RHR pump remains in shutdown cooling mode of operation.
The "B", "C" and "D" RHR pumps are injecting into the reactor vessel.

D. RHR loop suction and discharge valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO remain open.
The "A" RHR pump remains in shutdown cooling mode of operation.
The "B", "C" and "D" RHR pumps are not injecting into the reactor vessel.

---

**Question Data**

**Answer: A** RHR loop suction and discharge valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A" & "C" RHR pump breakers are pumping (cycling open/closed).
The "B" and "D" RHR pumps are running but not injecting into the reactor vessel.

Need to get injection?

**Explanation/Justification:**

what does the pump need to get injection?

**A.** correct answer
**B.** no injection occurs without operator intervention
**C.** F008, F009, and F015 auto close below +13" reactor water level.
**D.** F008, F009, and F015 auto close below +13" reactor water level.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **205000** | **Shutdown Cooling System (RHR Shutdown Cooling Mode)** | **Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM/MODE including:** | **Pump trips** |

okay - pumps will trip based on valve out of cycling on logic

K/A# **205000.A3.02** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-149-002, TM-OP-049**

Question Source: **Modified** **Limerick 2, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

31 During a LOCA, HPCI automatically initiated, then tripped. The operator notes the following indications:

| | |
|---|---|
| - Turbine Stop Valve | Closed |
| - HPCI Turbine RPM | Zero |
| - HPCI TURB TRIPPED | Alarm Sealed In |
| - HPCI TURB TRIP SOLENOID ENERG | Has NOT alarmed |

What caused the HPCI turbine to trip?

A. loss of oil pressure.

B. high exhaust pressure.

C. high reactor water level.

D. low steam supply pressure.

---

**Question Data**

**Answer: A** loss of oil pressure.

**Explanation/Justification:**

**A.** Correct answer, Turb Tripped alarm (AR-114-A01) is from valve position, trip solenoid alarm is direct turbine trip, all other distracters are direct turb trips.

**B.** Direct turb trip would cause trip solenoid alarm

**C.** Direct turb trip would cause trip solenoid alarm

**D.** Direct turb trip would cause trip solenoid alarm

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **206000.2.4.10** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-114-001**

Question Source: **Modified** **Brunswick, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

Generic K/A will be supplied

Answer is not easy distractors are plant-specific valid

# SSES LOC 19 NRC Exam

32 The Core Spray Loop "A" initiation logic channel has experienced a loss of power from 125 VDC Class 1E Bus A (1D614).

Which group of alarms from 1C601 Alarm Panel (AR-109-001) and BIS Display for Core Spray (AR-153-001) would be the expected response to the loss of the 1D614 power supply to the Core Spray Logic?

A. LOOP A OL OR POWER LOSS (AR-153-A01)
LOOP A OUT OF SERVICE (AR-153-B01)

B. LOOP A OL OR POWER LOSS (AR-153-A01)
LOOP A RELAY LGC PWR LOSS (AR-153-A03)

C. LOOP A RELAY LGC PWR LOSS (AR-153-A03)
CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

D. CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)
LOOP A OUT OF SERVICE (AR-153-B01)

Both Elect Power Supply too many alarms to be expected knowledge for Memory

**Question Data**

**Answer: C** LOOP A RELAY LGC PWR LOSS (AR-153-A03)
CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

**Explanation/Justification:**

**A.** Loss of 120VAC control power to any of following valves due to loss of associated power source or breaker racked out.
Thermal overload on any of above valves with CORE SPRAY LOOP A MOV OL BYPS HS-E21-1S12A in TEST.
74 Relay failure on any of above valves.
Loop A out of service caused by the handswitch on the vertical panel not a loss of logic power.

**B.** LOOP A OL OR POWER LOSS (A01)
Loss of 120VAC control power to any of following valves due to loss of associated power source or breaker racked out.
Thermal overload on any of above valves with CORE SPRAY LOOP A MOV OL BYPS HS-E21-1S12A in TEST.
74 Relay failure on any of above valves.

**C.** correct answer,

**D.** Loop A out of service caused by the handswitch on the vertical panel not a loss of logic power.

that what the AR Panel may be mentioned for! Valid L.O.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 209001 | Low Pressure Core Spray System | Knowledge of electrical power supplies to the following: | Initiation logic |

K/A# 209001.K2.03 K/A Importance 2.9 Exam Level RO

References provided to Candidate None Technical References: AR-153-001, AR-109-001

Question Source: New Level Of Difficulty: (1-5) 2

Question Cognitive Level: Memory 10 CFR Part 55 Content: 55.41

# SSES LOC 19 NRC Exam

33 How does the Automatic Depressurization System (ADS) logic determine that the Core Spray System is available prior to depressurizing the reactor?

A. Both Core Spray Pumps in one loop are operating at normal discharge pressure.

B. At least one Core Spray Pump breaker is closed and the associated loop minimum flow valve is closed.

C. Both Core Spray loop flow rates must be greater than 2000 gpm with the associated minimum flow valve closed.

D. At least one of the two Core Spray Pumps in one loop is operating at normal discharge pressure.

---

**Question Data**

**Answer: A** Both Core Spray Pumps in one loop are operating at normal discharge pressure.

**Explanation/Justification:**

**A.** Correct answer
**B.** Breaker closure is not a permissive.
**C.** Flow is not a permissive
**D.** Both pumps to be running with normal discharge pressure.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **209001** | **Low Pressure Core Spray System** | **Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following:** | **ADS logic** |

K/A# **209001.K3.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-83E**

Question Source: **Modified** **FITZPATRICK, 1993** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

34 An ATWS transient occurs and boron must be injected with the Standby Liquid Control System (SLC).

Which of the following describes the effect of a successful initiation of SLC on these indications:

1) Squib Valve Ready (Continuity) lights
2) Alarm on Panel 1C601, SBLC SQUIB VALVES LOSS OF CKT CONTINUITY AR-107-A03
3) Pump discharge pressure

A. 1) Illuminated
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

B. 1) Extinguished
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

C. 1) Illuminated
2) Annunciator not in Alarm.
3) Just above reactor pressure

D. 1) Extinguished
2) Annunciator not in Alarm
3) Just above reactor pressure

---

**Question Data**

**Answer: B** 1) Extinguished
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

**Explanation/Justification:**

**A.** Continuity lights go out.
**B.** correct answer, Squib valve fires, causing loss of continuity, alarm annunciates indicating loss of continuity, pumps start with discharge pressure slightly greater than reactor pressure.
**C.** Continuity lights go out. Alarm is annunciated, discharge pressure 200 psig reactore pressure.
**D.** Alarm is annunciated, discharge pressure 200 psig above reactor pressure.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **211000** | **Standby Liquid Control System** | **Ability to manually operate and/or monitor in the control room:** | **System initiation: Plant-Specific** |

K/A# **211000.A4.08** K/A Importance **4.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-153-001, TM-OP-053**

Question Source: **Modified** **Duane Arnold 1, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

35 Given the following conditions:

- Unit 2 has experienced a failure-to-scram (ATWS)
- The Standby Liquid Control (SLC) system was initiated and injected for 52 minutes at normal flowrate before both SLC Pumps failed
- Reactor power is in the Source Range

How does this failure affect the planned reactor cooldown and depressurization?

A. Reactor Engineering must make the determination if current boron concentration will allow a complete cooldown.

B. Cooldown can be accomplished if completed before Xenon decays out of the core.

C. Reactor boron concentration is sufficient to allow a complete cooldown with a maximum of 8 control rods not fully inserted.

D. Reactor boron concentration is sufficient to allow a complete cooldown under any plant conditions.

Too many Bank type questions

**Question Data**

**Answer: D** Reactor boron concentration is sufficient to allow a complete cooldown under any plant conditions.

**Explanation/Justification:**

**A.** not required.
**B.** CSBW will account for Xe decay as well.
**C.** CSBW will handle any number of rods out
**D.** correct answer, 2 pumps at TS minimum of 41.2 gpm for 52 minutes is 4284 gallons, greater than the CSBW of 4191 gallons.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **211000.2.1.7** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-113**

Question Source: **Modified** **SSES, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

LOD 1-2

ok

Generic KA needed

Another Basic

# SSES LOC 19 NRC Exam

36 During plant shutdown, which of the following Reactor Protection System automatic scrams is bypassed by taking the Mode Switch from RUN to STARTUP ?

A. Scram Discharge Volume Level High.

B. Main Steam Isolation Valve closure

C. Turbine Control Valve Fast Closure.

D. Turbine Stop Valve Closure.

---

**Question Data**

**Answer: B** Main Steam Isolation Valve closure

**Explanation/Justification:**

**A.** not bypassed in startup
**B.** correct answer
**C.** bypassed by turbine 1st stage pressure
**D.** bypassed by turbine 1st stage pressure

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **212000** | **Reactor Protection System** | **Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following:** | **Main steam system** |

K/A# **212000.K1.14** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-58**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) ~~3~~

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

Elect Supply

37 With the plant is operating at 80% power, a loss of power to MCC 1B217 occurrs.

Which of the below are expected alarms?

A. RPS CHANNEL B1/B2 AUTO SCRAM (AR-104-A01), NEUTRON MON CHAN B SYSTEM TRIP (AR-104-A04), and CORE SPRAY LOOP B OUT OF SERVICE ( AR-113-B02 )

B. RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP B OUT OF SERVICE ( AR-113-B02 )

C. RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP A OUT OF SERVICE ( AR-109-B02 )

D. RPS CHANNEL B1/B2 AUTO SCRAM (AR-104-A01), NEUTRON MON CHAN B SYSTEM TRIP (AR-104-A04)

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**Question Data**

**Answer: C** RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP A OUT OF SERVICE ( AR-109-B02 )

**Explanation/Justification:**

**A.** Wrong division
**B.** Core spray valves FO04 and 05 powered from 1B217
**C.** correct answer,
**D.** Wrong division

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **212000** | **Reactor Protection System** | **Knowledge of electrical power supplies to the following:** | **RPS motor-generator sets** |

K/A# **212000.K2.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-058**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

Is this rejd Probably for memory

# SSES LOC 19 NRC Exam

38 A reactor startup is in progress.

- Power is on Range 2 of the IRM's.
- The "A" IRM increases to 124/125 of scale and the "F" IRM increases to 123/125 of scale.

A Division I half scram occurs.

Which of the following describes the unit RO expected actions?

A. Range up on "A" IRM and reset the Division I side half scram, continue the startup

B. Range up on "F" IRM and continue the startup

C. Place the reactor Mode Switch in Shutdown.

D. Range up on "A" and "F" IRM, reset the Division I side half scram continue the startup

---

**Question Data**

**Answer: C** Place the reactor Mode Switch in Shutdown.

**Explanation/Justification:**

**A.** a full scram should have occurred for these conditions
**B.** 'F' IRM already above trip setpoint
**C.** correct
**D.** a full scram should have occurred for these conditions

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **215003** | **Intermediate Range Monitor (IRM) System** | **Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including:** | **RPS status** |

K/A# **215003.A3.03** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-AD-001**

Question Source: **Modified** / **Limerick 1, 1997** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

H

# SSES LOC 19 NRC Exam

39 Which of the following describes how a gradual DECREASE in Argon fill gas pressure will affect the response of a Source Range Monitoring (SRM) detector?

CFR?

A. Gamma sensitivity would Decrease
Neutron sensitivity would Decrease

B. Gamma sensitivity would NOT change
Neutron sensitivity would NOT change

C. Gamma sensitivity would Increase
Neutron sensitivity would NOT change

D. Gamma sensitivity would NOT change
Neutron sensitivity would Decrease

---

**Question Data**

**Answer: A** Gamma sensitivity would Decrease
Neutron sensitivity would Decrease

**Explanation/Justification:**

**A.** Correct answer

**B.** Fission causes recoil within the detector chamber and ionize the Argon gas, decrease in argon gas will reduce sensitivity##Gamma causes direct ionization of argon, decrease in argon gas will reduce sensitivity

**C.** Fission causes recoil within the detector chamber and ionize the Argon gas, decrease in argon gas will reduce sensitivity.

**D.** Gamma causes direct ionization of argon, decrease in argon gas will reduce sensitivity

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **215004** | **Source Range Monitor (SRM) System** | **Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM:** | **Detector operation** |

K/A# **215004.K5.01** K/A Importance **2.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-078**

Question Source: **Modified** **Hope Creek Unit 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

(work to make more complicated / relevant

# SSES LOC 19 NRC Exam

40 - A Unit 2 startup is in progress with power at 20%
- Recirculation flow is 30%
- The "A" APRM Flow Unit output remains at 30% as recirculation flow is raised

As the plant startup continues, what will be the FIRST protective action to occur and the reason for that action?

spell out

A. A half scram will occur due to a flow unit "inop" signal

B. A control rod block will occur due to a flow biased neutron flux upscale

C. A control rod block will occur due to a flow unit comparator trip.

D. A full scram will occur due to a flow biased neutron flux upscale

**Question Data**

**Answer: C** A control rod block will occur due to a flow unit comparator trip.

**Explanation/Justification:**

**A.** not a half scram signal
**B.** does not occur first
**C.** Correct, trip set at 10% difference between units
**D.** not the first event to occur.

Testing Effect on Control Rod System

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 215005 | **Average Power Range Monitor/Local Power Range Monitor System** | **Knowledge of the effect that a loss or malfunction of the following will have on the APRM/LPRM:** | **Flow converter/comparator network: Plant-Specific** |

K/A# **215005.K6.07** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-078**

Question Source: **Modified** **Peach Bottom, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

Too many Elect Supply

41 Unit 1 at 100% power.

125V DC PANEL 1L610 SYSTEM TROUBLE AR-106-A12 annunciates, indicating a loss of power to Division I 1D614.

How will a loss of Division I 1D614 impact auto initiation and/or steam leak detection isolation of RCIC?

A. Auto initiation will not function if needed. Steam leak detection will not function if needed.

B. Auto initiation will function if needed. Steam leak detection will function if needed by closing steam supply header inboard (HV-149-F007) and warmup isolation (HV-149-F088) valves.

C. Auto initiation will not function if needed. Steam leak detection will function if needed by closing steam supply header inboard (HV-149-F007) and warmup isolation (HV-149-F088) valves.

D. Auto initiation will not function if needed. Steam leak detection will function if needed by closing steam supply header outboard isolation valve (HV-149-F008).

**Question Data**

**Answer: C** Auto initiation will not function if needed. Steam leak detection will function if needed by closing steam supply header inboard (HV-149-F007) and warmup isolation (HV-149-F088) valves.

**Explanation/Justification:**

**A.** Div II of steam leak detection isolation will function, closing inboard isolation valves.
**B.** Auto initiation will not function.
**C.** Correct answer.
**D.** Div II of steam leak detection isolation will function, not Div I.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **217000** | **Reactor Core Isolation Cooling System (RCIC)** | **Knowledge of electrical power supplies to the following:** | **RCIC initiation signals (logic)** |

K/A# **217000.K2.02** K/A Importance **2.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-050**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

42 The following plant conditions exist:

- A reactor transient is in progress.
- 'B' loop of RHR is in operation.
- 'A' Core Spray pump in operation
- Drywell pressure 2.5 psig.
- Reactor water level -140 inches.

The following alarm conditions are noted:

RX LO LEVEL SIGNAL A CONFIRMED AR-110-B01 - Alarm window Not lit
RX LO LEVEL SIGNAL B CONFIRMED AR-110-B03 - Alarm window Lit
ADS LOGIC C & D TIMER INITIATED AR-110-A03 & AR-110-A04 alarms came in 40 seconds ago.

Which of the following states the response of the ADS to plant conditions with no operator action?

A. 6 ADS valves will open in approximately 60 seconds from Div I & II logic.

B. 3 ADS valves will open in approximately 60 seconds from Div II logic.

C. 3 ADS valves will open in approximately 520 seconds.

D. 6 ADS valves will open in approximately 60 seconds from Div II logic.

---

**Question Data**

**Answer: D** 6 ADS valves will open in approximately 60 seconds from Div II logic.

**Explanation/Justification:**

**A.** Div I does not have +13 " confirmation signal nor Divl pp permissive
**B.** Div II will initate all ADS valves.
**C.** Drywell pressure does not require to be bypassed with 7 minute timer.
**D.** Correct answer, Div I does not have +13 " confirmation signal, Div II will initate all ADS valves.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **218000** | **Automatic Depressurization System** | **Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM:** | **Nuclear boiler instrument system (level indication)** |

K/A# **218000.K6.03** K/A Importance **3.8** Exam Level **RO**
References provided to Candidate **None** Technical References: **TM-OP-83**
Question Source: **Modified** **Hope Creek Unit 1, 1995** Level Of Difficulty: (1-5) **3**
Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

43 Which of the statements below is correct regarding the arming and de-pressing of the four (4) NSSSS pushbuttons on panel 1C651.

A. "A" and "C" pushbuttons will cause a full MSIV isolation.

B. "A" and "B" pushbuttons will cause a full inboard isolation including all 8 MSIVs.

C. "B" and "C" pushbuttons will cause an inboard MSIV isolation.

D. "A" and "D" pushbuttons will cause an outboard MSIV isolation.

---

**Question Data**

**Answer: B** "A" and "B" pushbuttons will cause a full inboard isolation including all 8 MSIVs.

**Explanation/Justification:**

**A.** cause no isolation
**B.** correct answer
**C.** 'B' & 'C' button causes an inboard isolation, inboard and outboard MSIV isolation
**D.** 'A' & 'D' causes a full inboard and outboard isolation

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **223002** | **Primary Containment Isolation System/Nuclear Steam Supply Shut-Off** | **Knowledge of PCIS/NSSSS design feature(s) and/or interlocks which provide for the following:** | **Manual initiation capability: Plant-Specific** |

K/A# **223002.K4.03** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-059**

Question Source: **Modified** **SUSQUEHANNA, 1989** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

44 With the reactor at 100% power, which of the following would be an indication of an open Safety Relief Valve (SRV)?

A. Total indicated steam flow increase.

B. Indicated Feed flow less than Steam flow. Not Plausible

C. SRV Tailpipe temperature stable at 500 deg F.

D. Reactor thermal power increase.

---

**Question Data**

**Answer: D** Reactor thermal power increase.

**Explanation/Justification:**

**A.** Indicated total steam flow will decrease.
**B.** Feed flow will indicate greater than steam flow.
**C.** Tail pipe temperature will peak at approximately 320 degrees.
**D.** correct answer, power will increase slightly due to reduced feed heating due to loss of extraction steam heating.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **239002** | **Relief/Safety Valves** | **Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including:** | **Reactor power** |

K/A# **239002.A1.06** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-183-001**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

45 Unit 1 is at 84% power with Feedwater Control in 'Average Level Control' and Three Element Control.

What would be the response if the Unit 1 "A" Narrow Range Level instrument were equalized.(assume no operator action)

A. All three feedpumps and the main turbine trip.

B. RPV level will lower to approximately 23 inches and remain steady at the new lower level.

C. Division I scram on low level, High level for Division II.

D. RPV level will rise to approximately 47 inches and remain steady at the new higher level.

---

**Question Data**

**Answer: B** RPV level will lower to approximately 23 inches and remain steady at the new lower level.

**Explanation/Justification:**

**A.** signal only affected one instrument, need two out of three to trip turbines
**B.** correct answer, zero delta p is a failure upscale.
**C.** RPV level will not lower to scram setpoint because of average level circuit
**D.** the zero DP signal will simulate a high level signal and cause FW control to lower level not control higher.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **259002** | **Reactor Water Level Control System** | **Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** | **Loss of reactor water level input** |

K/A# **259002.A2.03** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-045**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

Basic System Knowledge

46 The following conditions exist:
- Unit 2 is operating at 100% power.
- The Standby Gas Treatment System (SGTS) is in the standby lineup.
- A valid Unit 2 high drywell pressure initiation signal occurs.

Which of the following are the locations from which the SGTS will automatically take suction.

A. U-2 drywell, Reactor Building Zone 2 and Reactor Building Zone 3.

B. Reactor Building Zone 2 and Reactor Building Zone 3, Unit 2 HPCI Barometric Condenser.

C. U-2 drywell and U-2 HPCI Barometric Condenser.

D. Only the U-2 Reactor Building Zone 2 and Reactor Building Zone 3.

---

**Question Data**

**Answer: B** Reactor Building Zone 2 and Reactor Building Zone 3, Unit 2 HPCI Barometric Condenser.

**Explanation/Justification:**

**A.** U-2 Drywell does not align automatically
**B.** correct answer
**C.** U-2 Drywell does not align automatically
**D.** HPCI Barametric Condenser exhausts to the SBGTS suction.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **261000** | **Standby Gas Treatment System** | **Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following:** | **Primary containment pressure** |

K/A# **261000.K1.11** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-073**

Question Source: **Modified** **Browns Ferry 1, 2, & 3, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

47 Which of the following ESS 4 KV bus feeder breaker trips will NOT auto transfer the bus to its alternate source?

A. Incoming feeder overcurrent

B. ESS Transformer pressure relay.

C. Undervoltage

D. ESS Transformer differential current

**Question Data**

**Answer: A** Incoming feeder overcurrent

**Explanation/Justification:**

**A.** correct answer,
**B.** transformer lockout signal only
**C.** causes transfer to alternate source
**D.** transformer lockout signal only

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **262001** | **A.C. Electrical Distribution** | **Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following:** | **Off-site power system** |

K/A# **262001.K3.05** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-004**

Question Source: **Modified** **Peach Bottom, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

48 1E Instrument AC UPS Distribution 1D130 supplies power to 1Y128 and 0Y301.

Which of the following is the correct electrical configuration for 1D130 to supply power to 1Y128 and 0Y301 immediately after the prefered source 1B246 is lost?

typo

A. 1B226, Alternate source stepped down to 120VAC

B. Battery, 1D133 inverted to 120VAC

C. 1B226, Maintenance Backup, rectified to 250VDC, inverted to 120VAC

D. 1B226, Maintenance Backup, rectified to 120VDC, inverted to 120VAC

---

**Question Data**

**Answer: B** Battery, 1D133 inverted to 120VAC

**Explanation/Justification:**

**A.** Battery is first alternate power supply.

**B.** Correct answer, battery, UPS being supplied from the dedicated battery can carry the distribution panels for approximately 20 minutes. When the external battery's output decreases to less than 210 VDC, the Static Transfer switch operates to supply alternate power to the distribution panels, and alternate supply is via 480-208/120 V step down transformers

**C.** Maintenance backup is used for maintenance and is a manual transfer.

**D.** Maintenance backup is used for maintenance and is a manual transfer.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **262002** | **Uninterruptable Power Supply (A.C./D.C.)** | **Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following:** | **Transfer from preferred power to alternate power supplies** |

K/A# **262002.K4.01** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-017**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

cueing: correct answer (B) is only "battery" choice

will make (A) a battery incorrect choice

# SSES LOC 19 NRC Exam

49 The Nuclear Plant Operator reports that both ground detection lights on a 250 V DC Battery Charger are dimmer than normal and that one light is brighter than the other.

What is the status of the 250 VDC bus?

A. Unable to determine ground exists based on indications provided.

B. A ground exists ONLY on the bus with the brighter light.

C. A ground exists ONLY on the bus with the dimmer light.

D. Grounds exist on both busses with the ground of the greater magnitude on the bus with the dimmer light.

---

**Question Data**

**Answer: D** Grounds exist on both busses with the ground of the greater magnitude on the bus with the dimmer light.

**Explanation/Justification:**

**A.** lights dim with grounds on both buses

**B.** bright light occurs with single ground on the ungrounded bus

**C.** dim light occurs with single ground

**D.** Correct answer If one light burns bright and the other light is dim or out, then a single ground has occurred on the side with the dimly lit light. If grounds are present on both buses then the side with the dimly lit light has the grounds of greater magnitude.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **263000** | **D.C. Electrical Distribution** | ***Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:*** | **Grounds** |

K/A# **263000.A2.01** K/A Importance **2.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

add (b) part of K/A to question
(ie use of procedure to fix ground, etc)
to improve LOD and address K/A

# SSES LOC 19 NRC Exam

50 During normal operation, the AC power supply to the 1D663 Battery Charger is lost.

Which of the following describe the automatic actions for the loss of the AC power to this battery charger?

A. The alternate battery charger immediately feeds the bus,
250V DC PANEL 1L650 SYSTEM TROUBLE AR-105-A11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

B. The battery charger is immediately supplied by alternate AC power,
250V DC PANEL 1L650 SYSTEM TROUBLE AR-105-A11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

C. The battery backfeeds the inverter and forward feeds the bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-105-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

D. The battery immediately feeds the 250 VDC bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-105-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

change all -105's to -106's

**Question Data**

**Answer: D** The battery immediately feeds the 250 VDC bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-105-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

Is this real knowledge for mem?

**Explanation/Justification:**

**A.** No alternate battery charger on Div II, only on Div I.
**B.** No alternate AC power supply for battery charger.
**C.** Battery feeds DC distribution bus directly.
**D.** correct answer, Control room alarm received from local alarm panel, local alrm panel alarm due to AC supply to battery charger lost.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **263000** | **D.C. Electrical Distribution** | **Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including:** | **Meters, dials, recorders, alarms, and indicating lights** |

K/A# **263000.A3.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-106-001**

Question Source: **Modified** **Browns Ferry 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

51 The A Diesel Generator (DG) is being run for the monthly operability check.

The PCOP has just closed the diesel output breaker to parallel the DG to the bus, when he notices the following:

- DG Frequency is 60 Hz
- DG Kilowatts is 5 Kw
- DG Kilovars is -4500 Kvar

Attempting to lower Kvars on the DG, the operator takes the SPEED ADJUST switch to lower. SELECT the DG response to this action:

ASK Spec Looks OKay

A. The DG will trip on field loss of voltage.

B. Frequency will decrease rapidly.

C. The DG will trip on reverse power.

D. The DG will slip a pole.

---

**Question Data**

**Answer: C** The DG will trip on reverse power.

**Explanation/Justification:**

**A.** trip on reverse power
**B.** Frequency can't change with D/G tied to grid
**C.** correct answer, D/G not loaded any decrease in fuel will reduce load further.
**D.** trip on reverse power not slip a pole

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **264000** | **Emergency Generators (Diesel/Jet)** | **Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including:** | **Maintaining minimum load on emergency generator (to prevent reverse power)** |

K/A# **264000.A1.09** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **SO-024-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

52 Maintenance is requesting that both Containment Instrument Gas (CIG) compressors be removed from service due to a common mode failure potential. The plan requires that Instrument Air be lined up to the CIG system for approximately 12 hours while the compressors are repaired.

What would be the ~~problem with~~ result of this plan if Instrument Air supply ~~to CIG were~~ is then lost?

Stem focus?

A. Pressure for the 90# CIG header from the storage bottles, no valves repositioning.

B. No pressure for the 90# CIG header, Closure of Inboard MSIVs, No pressure for the 150# CIG header, Safety Relief valves work only in Safety function.

C. No pressure for the 90# CIG header, Closure of Inboard MSIVs, loss of Drywell Cooling.

D. Pressure for the 90# and 150# CIG header from the backup Service Air tie , no valves repositioning.

---

**Question Data**

**Answer: C** No pressure for the 90# CIG header, Closure of Inboard MSIVs, loss of Drywell Cooling.

**Explanation/Justification:**

**A.** 150# header storgae bottles can not feed the 90# header.

**B.** 150# header would be supplied with the storage bottles for longer than 12 hours.

**C.** Correct answer. 90# header can not be supplied from Instrument air due to check valve arrangement.

**D.** There is not CIG/Service air cross tie, and supply to 90# header can not supply 150# header.

fix

It sounds like Instrument Air can't supply CIG completely because of check valve?

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **300000** | **Instrument Air System (IAS)** | **Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following:** | **Containment air system** |

K/A# **300000.K3.01** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-025**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

53 Unit 1 is at 100% power when a pipe break occurs on the Unit 1 TBCCW inlet to the in service heat exchanger.

What alarms and plant indications would be expected as a result of the pipe break?

A. TBCCW HEAD TANK HI-LO LEVEL AR-123-G06
TBCCW HX AREA FLOODED AR-123-H01
TBCCW Heat Exchanger Discharge Pressure indication (PI-14409) downscale
Off gas precoolers Hi temperature

B. TBCCW PUMPS DISCHARGE HEADER LO PRESS G03
TBCCW HEAT EXCHANGER HEADER LO PRESS AR-123-G04
Standby TBCCW pump running
Instrument Air Compressors tripped

C. TBCCW PUMPS DISCHARGE HEADER LO PRESS AR-123-G03
TBCCW HEAD TANK HI-LO LEVEL AR-123-G06
TBCCW HX AREA FLOODED (H01)
PASS Sample cooler Hi temperature

D. TBCCW HX AREA FLOODED AR-123-H01
TBCCW Heat Exchanger Discharge Pressure indication (PI-14409) downscale
Standby TBCCW pump running
RFPT Lube Oil Cooler Hi temperature

---

**Question Data**

**Answer: B** TBCCW PUMPS DISCHARGE HEADER LO PRESS G03
TBCCW HEAT EXCHANGER HEADER LO PRESS AR-123-G04
Standby TBCCW pump running
Instrument Air Compressors tripped

**Explanation/Justification:**

**A.** Offgas precooler RBCCW
**B.** Correct answer
**C.** Pass cooler RBCCW
**D.** RFPT Lube Oil Cooler Service water

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **400000** | **Component Cooling Water System (CCWS)** | **Ability to manually operate and/or monitor in the control room:** | **CCW indications and control** |

K/A# **400000.A4.01** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-115-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

54 The reactor is at 100% power. CRD PUMP 1P132B is out of service. Annunciator "CRD PUMP `A' TRIP " alarms.

How is the movement of the control rods from the control room affected?

A. Rods can be scrammed and withdrawn, but not individually inserted.

B. Rods can be scrammed and inserted, but not individually withdrawn.

C. Rods can only be scrammed.

D. Scram Accumulator pressure assures that all functions work, although rod motion is slow, and CRD mechanism cooling is lost.

---

**Question Data**

**Answer: C** Rods can only be scrammed.

**Explanation/Justification:**

**A.** No insert or withdraw available
**B.** No insert or withdraw available
**C.** correct answer
**D.** accumulator function is for scram only

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **201001** | **Control Rod Drive Hydraulic System** | **Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System:** | **A.C. power** |

K/A# **201001.K6.05** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-155-007**

Question Source: **Modified** **Duane Arnold, 1996** Level Of Difficulty: (1-5)

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

bet up plausibility of distractors

# SSES LOC 19 NRC Exam

55 Select the method by which reactivity insertion rate is regulated for control rod withdrawals and insertions.

Reactivity insertion rate is controlled by:

A. throttling the water flow entering and leaving the over-piston area of the drive mechanism.

B. automatically varying the position of the Control Rod Drive Hydraulic Pressure Control Valve.

C. automatically varying the operating position of the Control Rod Drive Hydraulic Flow Control Valve.

D. throttling the water flow entering and leaving the under-piston area of the drive mechanism.

Credible distractor

**Question Data**

**Answer: D** throttling the water flow entering and leaving the under-piston area of the drive mechanism.

**Explanation/Justification:**

**A.** over-piston area flow is leaving is not thottled.
**B.** pressure control valve is not automatically positioned.
**C.** flow control valve controls total system flow not regulated flow to each mechanism
**D.** correct answer

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **201003** | **Control Rod and Drive Mechanism** | **Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including:** | **CRD drive water flow** |

K/A# **201003.A1.03** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-055**

Question Source: **Modified** **River Bend 1, 1997** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

56 What is the design bases for allowing the Rod Sequence Control System to be automatically bypassed above 20% power?

A. No combination of Operator errors could result in fuel damage due to a Control Rod Drop Accident.

B. The power excursion caused by a Control Rod Drop Accident would be terminated by an APRM hi-flux SCRAM.

C. The RBM prevents any control rod from attaining the rod worth necessary to damage fuel on a Control Rod Drop Accident.

D. Rod Worth Minimizer wiil continue to enforce the rod control sequence.

---

**Question Data**

**Answer: A** No combination of Operator errors could result in fuel damage due to a Control Rod Drop Accident.

**Explanation/Justification:**

**A.** correct answer, voids in large enough quantity to minimize differential rod worths.
**B.** possibly but not the basis
**C.** RBM minimizes local power increase by limiting amount of power change, does not affect rod worth
**D.** RWM also bypassed

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **201004** | **Rod Sequence Control System (Plant Specific)** | **Knowledge of the operational implications of the following concepts as they apply to ROD SEQUENCE CONTROL SYSTEM:** | **Prevention of clad damage if a control rod drop accident (CRDA) occurs: BWR-4, 5** |

K/A# **201004.K5.01** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-056**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

57 With the Rod Worth Minimizer keylock in NORMAL, a loss of rod position signal from the selected rod to the Rod Worth Minimizer will cause:

A. only a withdraw block if power is less than the Low Power Set Point.

B. only a "SYSTEM ERROR" display.

C. a withdraw and insert rod block if power is less than the Low Power Set Point.

D. a withdraw and insert rod block at any power.

---

**Question Data**

**Answer: C** a withdraw and insert rod block if power is less than the Low Power Set Point.

**Explanation/Justification:**

**A.** also insert block
**B.** withdraw and insert block
**C.** correct answer
**D.** blocks are bypassed above 20% power

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **201006** | **Rod Worth Minimizer System (RWM) (Plant Specific)** | **Ability to predict and/or monitor changes in parameters associated with operating the ROD WORTH MINIMIZER SYSTEM (RWM) controls including:** | **Rod position: P-Spec(Not-BWR6)** |

K/A# **201006.A1.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-031D**

Question Source: **Modified** **Susquehanna 1 & 2, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

58 A Design Basis Accident has occurred on Unit 1 with a lockout on the 1A201 ESS bus.

Which of the following describes the ~~correct~~ impact and why? "Correct impact"?

A. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will not operate, power is lost from 1B219.

B. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with power from 1B219.

C. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with alternate power from 1B229.

D. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will not operate power is lost from 1B229.

---

**Question Data**

**Answer: B** RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with power from 1B219.

**Explanation/Justification:**

**A.** ATS will supply power from 1B230.

**B.** Correct answer, ESS Load Center 1B210 is normal power supply via Swing Bus MG Set for 1B219, loss of power from ESS bus 1A201 to 1B210 will cause ATS to seek power from 1B230.
The 1B219 alternate source is ESS Load Center 1B230

**C.** 1B229 is power supply for Div 2 valves.

**D.** 1B229 is power supply for Div 2 valves.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **202001** | **Recirculation System** | **Knowledge of electrical power supplies to the following:** | **Recirculation system valves** |

K/A# **202001.K2.03** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-004**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

59 While making a tour of the Unit 1 Lower Relay Room, you notice an alarm light on a RBM channel on top of panel 1C608. The light is labeled, "REF DNSCL".

What RBM function is associated with this alarm?

A. Automatically bypasses RBM

B. Initiates Rod Block

C. Bypasses Rod Insert Blocks

D. Indication only for the reference APRM at 30% power

---

**Question Data**

**Answer: A** Automatically bypasses RBM

**Explanation/Justification:**

**A.** correct
**B.** Not a RBM rod block
**C.** RBM does not block rod insertion
**D.** provides control function as well as indication.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **215002** | **Rod Block Monitor System** | **Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including:** | **Alarm and indicating lights: BWR-3, 4, 5** |

K/A# **215002.A3.03** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-078**

Question Source: **Modified** **Dresden 2 & 3, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

60 Unit 1 is operating at 100% power.

- Wide Range level indicates +50 inches.
- RX WATER HI LEVEL AR-101-A17 alarm on 1C651
- Green light above the LEVEL LOGIC RESET A HS-C32-1S04A switch is lit on 1C651.
- Green light above the LEVEL LOGIC RESET B HS-C32-1S04B switch is lit on 1C651.
- Green light above the LEVEL LOGIC RESET C HS-C32-1S04C switch is NOT lit on 1C651.

Which of the following is the expected response for the above indications

A. No Reactor Feedpumps tripped, no HPCI or RCIC trip alarms.

B. 'A', 'B' & 'C' Reactor Feedpumps tripped, no HPCI or RCIC trip alarms.

C. 'A' & 'B' Reactor Feedpumps tripped, 'C' Reactor Feed Pump feeding, trip alarms annunciated for HPCI or RCIC

D. No Reactor Feedpumps tripped, trip alarms annunciated for HPCI or RCIC

**Question Data**

**Answer: B** 'A', 'B' & 'C' Reactor Feedpumps tripped, no HPCI or RCIC trip alarms.

**Explanation/Justification:**

**A.** Trip logic for RFPs is actuated.
**B.** correct answer, two out of three logic for trip of feedpumps, HPCI-RCIC hi level trip come from different level switches thus possible not tripped.
**C.** No other alarms as mentioned in stem, thus no HPCI/RCIC alarms
**D.** RFPs tripped, no alarms annunciated for HPCI/RCIC

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **216000** | **Nuclear Boiler Instrumentation** | **Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following:** | **Protection against filling the main steam lines from the feed system** |

K/A# **216000.K4.09** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-080**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

61 Suppression Pool cooling is in service on Unit 2 with the 2A RHR pump in service when an electrical fault causes Bus 2A201 to deenergize.

All other buses remain energized.

Which of the following describes how Suppression Pool cooling will be affected by this bus loss?

A. The 2D RHR Pump will be deenergized; Loops A & B are available for Suppression Pool Cooling.

B. The 2A and 2C RHR Pumps will be deneergized; Loop B is available for Suppression Pool Cooling.

C. The 2A RHR Pump will be deenergized; Loop B is available for Suppression Pool Cooling.

D. The 2B and 2D RHR Pumps will be deneergized; Loop A is available for Suppression Pool Cooling.

**Question Data**

**Answer: C** The 2A RHR Pump will be deenergized; Loop B is available for Suppression Pool Cooling.

**Explanation/Justification:**

**A.** only '2A' pump not available.
**B.** only '2A' pump not available.
**C.** correct answer, '2A' pump powered from 2A ESS bus.
**D.** only '2A' pump not available.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **219000** | **RHR/LPCI: Torus/Suppression Pool Cooling Mode** | **Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE controls including:** | **System flow** |

K/A# **219000.A1.02** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-049, ON-204-210**

Question Source: **Modified** **Quad Cities 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: ~~Comprehension~~ F 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

62 Following an Auxiliary Bus Load Shed with all of the emergency busses energized, which of the following components will require operator actions to provide a source of cooling water?

A. Diesel Generators

B. Condensate pumps

C. HPCI room coolers

D. RHR pump room coolers

need distractor that has lost cooling water (in common w/ condensate) and requires new action

**Question Data**

**Answer: B** Condensate pumps

**Explanation/Justification:**

**A.** No additional operator actions required.
**B.** Correct answer, TBCCW available but will need ESW lined up to substitute for Service water.
**C.** No additional operator actions required.
**D.** No additional operator actions required.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **256000** | **Reactor Condensate System** | **Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** | **Loss of equipment component cooling water systems** |

K/A# **256000.A2.12** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-104-001**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

63 Unit 1 & 2 are operating at 100% power when the following annunciator is received in the Control Room;

REFUEL FLOOR WALL EXHAUST HI-HI RADIATION (AR-101-A05)

Which of the following is the expected ventilation response?

A. Isolation of Reactor Building Zone III Ventilation
Automatic start of both SGTS fans
RB Zone III Filtered Exh Fans 1V217A(B) and 2V217A(B) start.

B. Isolation of Reactor Building Zone III Ventilation
Automatic start of both SGTS fans
Automatic start of Reactor Building Zone III Recirculation System

C. SGTS Train A or B start.
Emergency Outside Air Supply Fan A(B) starts.
RB Recirc System to SBGT HD07543A(B) closes.

D. SGTS Train A or B start.
RB Zone III Iso Dampers HD27502A(B), HD27514A(B) HD27564A(B), HD17502A(B), HD17514A(B) and HD17564A(B) closes.
RB Zone III Filtered Exh Fans 1V217A(B) and 2V217A(B) start.

---

**Question Data**

**Answer: B** Isolation of Reactor Building Zone III Ventilation
Automatic start of both SGTS fans
Automatic start of Reactor Building Zone III Recirculation System

**Explanation/Justification:**

**A.** RB Zone III Filtered Exh Fans trip not start.
**B.** Correct answer
**C.** RB Recirc System to SBGT HD07543A(B) open instead of closing.
**D.** RB Zone III Filtered Exh Fans trip not start.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **272000** | **Radiation Monitoring System** | **Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including:** | **Ventilation system isolation indications** |

K/A# **272000.A3.06** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-079**

Question Source: **Modified** **Big Rock Point 1, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

64 A loss of Zone I and III ventilation has occurred with Unit 1 at 100% power. The loss of Reactor Building Ventilation Off-Normal procedure has been entered, room cooling initiated and maintenance support requested.

Which area is expected to have the most rapid temperature increase:

A. ESS Switchgear rooms

B. HPCI Room

C. Main Steam Pipe Tunnel

D. RWCU Pump Room

---

**Question Data**

**Answer: D** RWCU Pump Room

**Explanation/Justification:**

**A.** ESS Switchgear rooms has its own cooling system which remains in service with a loss of Zone I HVAC.
**B.** HPCI Room has it's own cooling system which remains in service with a loss of Zone I HVAC.
**C.** Main Steam Pipe Tunnel has it's own cooling system which remains in service with a loss of Zone I HVAC.
**D.** Correct answer

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **288000** | **Plant Ventilation Systems** | **Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on following:** | **Reactor building temperature: Plant-Specific** |

K/A# **288000.K3.02** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-134-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

65 Unit 1 had a loss of Drywell Cooling while operating at 100% power due to a spurious high Drywell pressure signal.

The high Drywell pressure signal is cleared, what needs to be completed to reset Drywell Cooling isolation logic?

typo

A. Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Go to open for A & B Drywell Cooler inboard and outboard isolation valves

B. Go to close for A & B Drywell Cooler inboard and outboard isolation valves
Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681

C. Reset MN STM LINE DIV I & II logic on 1C601
Go to close for A & B Drywell Cooler inboard and outboard isolation valves
Reset Div I & II RBCW ISO VALVE POS logic on 1C681

D. Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Verify A & B Drywell Cooler inboard and outboard isolation valves open

---

**Question Data**

**Answer: D** Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Verify A & B Drywell Cooler inboard and outboard isolation valves open

**Explanation/Justification:**

**A.** No need to open valves
**B.** No need to go to close.
**C.** No need to go to close
**D.** Correct answer

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **290001** | **Secondary Containment** | **Ability to manually operate and/or monitor in the control room:** | **System reset: Plant-Specific** |

K/A# **290001.A4.11** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-159-002**

Question Source: **Modified** **Hope Creek Unit 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

66 The US, PCO, and I&C have discussed the performance of a Surveillance Procedure on the ADS system. The annunciators expected to be received during the surveillance have been identified and reviewed.

Which of the following describes the required actions when one of the expected annunciators is received?

The operator shall acknowledge the alarm and:

A. is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.

B. is NOT required to report the alarm to the US. The associated alarm response procedure shall be checked.

C. the annunciator shall be reported to the US. The operator does NOT have to refer to the associated alarm response procedure.

D. the annunciator shall be reported to the US. The associated alarm response procedure shall be checked.

---

**Question Data**

**Answer: A** is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.

**Explanation/Justification:**

**A.** correct answer, The operator shall acknowledge the alarm and is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.
**B.** The Alarm Procedure does not have to be referred to under these conditions
**C.** No report is required to the US
**D.** No report is required to the US

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Conduct of Operations** | **Knowledge of operator responsibilities during all modes of plant operation.** |

K/A# **2.1.2** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-AD-004 (Sect 11.2.c)**

Question Source: **Modified** **Cooper 1, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

67 In accordance with STANDARDS OF SHIFT OPERATION, OP-AD-002, which of the following is the PREFERRED method to perform a required verification on a throttled valve set at two turns open?

A. Independent visual check of required valve position.

B. Perform a second valve operation to verify the position.

C. Observe flow indication through the throttled valve's system during system lineup.

D. Observe the initial operator's action in positioning the throttled valve.

answer almost common sense

NLO knowledge level

LOD = 1 does not ~~depth~~ discern

**Question Data**

**Answer: D** Observe the initial operator's action in positioning the throttled valve.

**Explanation/Justification:**

**A.** *throttle valves are aligned using concurrent verification*
**B.** throttle valves are aligned using concurrent verification
**C.** throttle valves are aligned using concurrent verification
**D.** correct answer, throttle valves are aligned using concurrent verification

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Conduct of Operations** | **Ability to perform specific system and integrated plant procedures during different modes of plant operation.** |

K/A# **2.1.23** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-AD-002 (7.3.5)**

Question Source: **Modified** **Quad Cities 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

new question

# SSES LOC 19 NRC Exam

68 During a startup on Unit 2 reactor, The Plant Control Operator withdraws control rod 26-27 from notch 32 to notch 34, reactor period changes from 200 seconds to a stable 50 second period.

Which of the following identifies the required action to be taken?

A. Re-insert control rods as necessary to achieve sub-criticality.

B. Shutdown the reactor until a thorough assessment has been performed.

C. Re-insert control rod 26-27 to obtain a stable period indication of greater than 100 seconds.

D. Do not move any additional rods until a Core Monitor is run.

---

**Question Data**

**Answer: C** Re-insert control rod 26-27 to obtain a stable period indication of greater than 100 seconds.

**Explanation/Justification:**

**A.** obtain a stable period indication of greater than 100 seconds.
**B.** obtain a stable period indication of greater than 100 seconds.
**C.** correct answer
**D.** obtain a stable period indication of greater than 100 seconds.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Conduct of Operations** | **Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.** |

K/A# **2.1.7** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **GO-200-002, 6.23**

Question Source: **Modified** **Browns Ferry 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **~~Comprehension~~** F 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

69 Given the following conditions:

- Reactor power is 40%
- ALL Turbine Bypass Valves fail OPEN.
- The MSIVs FAIL to automatically close.
- The MSIVs are closed manually

Prior to MSIV closure, which of the following combinations of reactor power and reactor pressure would indicate a safety limit violation had occurred?

A. Reactor power is 30% and RPV pressure is 810 psig

B. Reactor power is 30% and RPV pressure is 775 psig

C. Reactor power is 20% and RPV pressure is 795 psig

D. Reactor power is 10% and RPV pressure is 810 psig

---

**Question Data**

**Answer: B** Reactor power is 30% and RPV pressure is 775 psig

**Explanation/Justification:**

**A.** Pressure and flow within Safety Limit
**B.** correct answer, <785 psig, <10E6 lbm core flow, thermal power must be < 25%.
**C.** Power within Safety Limit of 25%
**D.** Pressure, power and flow within Safety Limit.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of limiting conditions for operations and safety limits.** |

K/A# **2.2.22** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **T.S. 2.1**

Question Source: **Modified** **Dresden 2, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

70 A Procedure Change for immediate use is required for RCIC Quarterly Flow Verification, SO-150-002. The HV-149-F022 TEST LINE ISO TO CST needs to be set to a position other than 40% OPEN as required by the procedure because of a pump impeller modification.

The Procedure Change requires the following steps to be completed prior to use of the procedure.

A. Obtain PORC committee review, Log PCAF in the control room Procedure PCAF Log, Stamp PCAF placed in controlled manuals as CONTROLLED, Deliver the original PCAF to training.

B. Obtain responsible Functional Unit Manager approval, Stamp PCAF placed in controlled manuals as CONTROLLED, QC approve insertion into appropriate manuals.

C. Log PCAF in the control room Procedure PCAF Log, Stamp PCAF placed in controlled manuals as CONTROLLED, Deliver the original PCAF to DCS, Obtain PORC committee review,.

D. Obtain responsible Functional Unit Manager approval, Stamp PCAF placed in controlled manuals as CONTROLLED, Deliver the original PCAF to DCS.

**Question Data**

**Answer: C** Log PCAF in the control room Procedure PCAF Log, Stamp PCAF placed in controlled manuals as CONTROLLED, Deliver the original PCAF to DCS, Obtain PORC committee review,.

**Explanation/Justification:**

**A.** PCAF is delivered to DCS.
**B.** QC does not approve PCAFs
**C.** Correct answer.
**D.** PCAF requires PORC approval.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of the process for controlling temporary changes.** |

K/A# 2.2.11 K/A Importance 2.5 Exam Level RO

References provided to Candidate **None** Technical References: **NDAP-QA-0002 (8.3)**

Question Source: **Modified** **Quad Cities 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

71 The plant is in MODE 4 preparing for a refueling outage.

What is/are the MINIMUM action(s) that must be performed to enter into MODE 5?

A. De-tension one reactor vessel head closure bolt.

B. Place the Reactor Mode Switch in the REFUEL position.

C. Place the Reactor Mode Switch in the REFUEL position and de-tension one reactor vessel head closure bolt.

D. Place the Reactor Mode Switch in the REFUEL position and de-tension all the reactor vessel head closure bolts.

---

**Question Data**

De-tension one reactor vessel head closure bolt.

**Explanation/Justification:**

**A.** correct answer, when detensioning of head begins, plant is in Mode 5 by definition of Tech Specs.
**B.** By procedure the mode switch should be in refuel, but head stud detensioning is accpted definition of Refuel Mode.
**C.** By procedure the mode switch should be in refuel, but head stud detensioning is accpted definition of Refuel Mode.
**D.** By procedure the mode switch should be in refuel, but head stud detensioning is accpted definition of Refuel Mode.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of refueling administrative requirements.** |

K/A# **2.2.26** K/A Importance **2.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **Tech Spec Definitions 1.01**

Question Source: **Modified** **Perry 1, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

replace question

# SSES LOC 19 NRC Exam

72 A 22 year old operator is working in a radiation field under the following conditions:

The operators cumulative dose for the year is 940 mrem.
The job is in a 20 mrem/hr radiation area.
No dose extension has been or will be authorized.

Select the number of hours the operator may work in the radiation area without exceeding the administrative limit for the year?

A. 26 hrs

B. 53 hrs

C. 203 hrs

D. 153 hrs

**Question Data**
**Answer: B** 53 hrs

**Explanation/Justification:**

**A.** 1000 Admin limit (minus) 940 cum dose = 60 dose available (divided by) 20 dose rate = 26 hours
**B.** correct answer, 2000 Admin limit (minus) 940 cum dose = 1060 dose available (divided by) 20 dose rate = 53 hours
**C.** 5000 NRC* (minus) 940 cum dose = 4060 dose available (divided by) 20 dose rate = 203 hours
**D.** 4000 Admin limit (minus) 940 cum dose = 1060 dose available (divided by) 20 dose rate = 153 hours

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Radiological Controls** | **Knowledge of 10 CFR 20 and related facility radiation control requirements.** |

K/A# **2.3.1** K/A Importance **2.6** Exam Level **RO**
References provided to Candidate **None** Technical References: **NDAP-QA-0625 (6.2)**
Question Source: **Modified** **Clinton 1, 2001** Level Of Difficulty: (1-5) **3**
Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

73 - Unit 1 is in Mode 3.
- It is desired to de-inert the Unit 1 Primary Containment as soon as possible to permit containment access for maintenance.

What flowpath is permited for de-inerting of the Unit 1 Suppression Chamber?

A. Both Standby Gas Treatment trains in service, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

B. One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

C. Both Standby Gas Treatment trains in service, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703, SUPP CHMBR VENT OB ISO HV-15704, and SUPP CHMBR VENT BYPS OB ISO HV-15705.

D. One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704, and SUPP CHMBR VENT BYPS OB ISO HV-15705.

**Question Data**

**Answer: B** One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

**Explanation/Justification:**

**A.** Only one SBGT train I/S at a time.
**B.** Correct answer.
**C.** Only one SBGT train I/S at a time, and either the 04 or 05 valve open, not both.
**D.** Either the 04 or 05 valve open, not both.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Radiological Controls** | **Knowledge of the process for performing a containment purge.** |

K/A# **2.3.9** K/A Importance **2.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-173-001**

Question Source: **Modified** **Quad Cities 1, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

74 EO-100-102, "RPV Control", directs the operator to reset the main generator lockout if RPV level can be maintained > -129 inches.

SELECT the correct bases for this step from the reasons listed below. Typo

A. To allow the Stator Water Cooling Pumps to be restarted to provide cooling to the main generator

B. To allow the Recirc Pumps to be restarted to establish forced reactor coolant circulation

C. To prevent a trip of the Stator Water Cooling Pump

D. To prevent a plant Auxiliary Bus load shed

---

**Question Data**

**Answer:** D To prevent a plant Auxiliary Bus load shed

**Explanation/Justification:**

**A.** No half scram signal received
**B.** No RRP runback signal exists
**C.** No Rod Block signal received
**D.** correct answer

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Emergency Procedures and Plan** | **Knowledge of the specific bases for EOPs.** |

K/A# **2.4.18** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-103-001**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

# SSES LOC 19 NRC Exam

75 The reactor is shutdown with one loop of shutdown cooling in use and NO Recirculation Pumps running.

How would the Shutdown Cooling System respond if Reactor Vessel Level decreased from +50 inches to +4 inches?

A. The shutdown cooling suction inboard and outboard isolation valves isolate, the operating RHR pump trips.

B. The shutdown cooling suction inboard and outboard isolation valves isolate, the operating RHR pump remains running.

C. Shutdown cooling continues unaffected.

D. The shutdown cooling suction inboard and outboard isolation valves isolate, the operating RHR pump trips, HV-151-F)15A opens, remaining RHR pumps auto start..

**Question Data**

**Answer: A** The shutdown cooling suction inboard and outboard isolation valves isolate, the operating RHR pump trips.

**Explanation/Justification:**

**A.** correct answer, running RHR pump will trip due to loss of suction path.
**B.** No suction path running pump will trip.
**C.** No suction path, running pump will trip.
**D.** F015 valve auto opens and RHR pumps auto start at -129.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Emergency Procedures and Plan** | **Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.** |

K/A# **2.4.9** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

assumes need only to tie them to a strategy

# SSES LOC 19 NRC Exam

76 Unit 2 is operating at power when a recirculation flow reduction event results in entry into Region 2 of the Power to Flow Map. Plant conditions PRIOR to the event were as follows:

- Reactor power was 90%.
- APRMs indicated 90%
- All LPRMs were above downscale alarms and below upscale alarms.
- Period meters indicated infinity.

After the flow reduction event and core flow first reaches its lowest flow rate, which of the following instrumentation responses would you use as the Unit Supervisor to justify entry into the core oscillation off normal procedure?

A. Peak to peak oscillations on RBM 10% and growing larger.

B. Period meters are oscillating and short period alarms are received on a 10 to 20 second frequency.

C. Peak to peak oscillations on APRM's 5% to 6% and their magnitude is growing larger.

D. Total Steam flow oscillations 10 to 12% and their magnitude is growing larger

**Question Data**

**Answer: C** Peak to peak oscillations on APRM's 5% to 6% and their magnitude is growing larger.

**Explanation/Justification:**

**A.** RBM not referenced in the off normal procedure.
**B.** No reference to period indication in the off normal procedure.
**C.** correct answer
**D.** Steam flow not referenced in the off normal procedure.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295001** | **Partial or Complete Loss of Forced Core Flow Circulation** | **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:** | **Neutron monitoring** |

K/A# **295001.AA2.02** K/A Importance **3.2** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-278-002**

Question Source: **Modified** **Peach Bottom, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

77 Unit 1 is operating at 100% power when an Instrument air line in the Turbine Building ruptures. The air compressors are unable to keep up with the loss of air and instrument air pressure is lowering.

What will the overall Reactor Pressure Vessel level control and pressure control strategy be for the loss of instrument air?

A. Pressure Control using Main Steam Line drains, condensate pumps for level control.

B. SRVs for pressure control, HPCI/RCIC for level control

C. Maximize CRD, RCIC for level control, SRVs for pressure control

D. HPCI/RCIC for level control, SRVs and Main Steam line drains for pressure control.

**Question Data**

**Answer: B** SRVs for pressure control, HPCI/RCIC for level control

**Explanation/Justification:**

**A.** Condenser is not available and no condensate line up is possible due to level control valves fail closed on a loss of air.

**B.** correct answer, Outboard MSIVs will go closed on a loss of air, therefore no steam for feedpumps or use of the main condenser for decay heat. Condensate will be unavailable due to no feedpath on a loss of air.

**C.** CRD flow control valves fail closed on a loss of air,

**D.** Condenser is not available for pressure control

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295019** | **Partial or Complete Loss of Instrument Air** | **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:** | **Status of safety-related instrument air system loads (see AK2.1-AK2.19)** |

K/A# **295019.AA2.02** K/A Importance **3.7** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-118-001, EO-100-102**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

modify accident description to apply to answer choices

78 Which of the following describes the consequences if a fuel handling accident occurs and the Unit 1 Refuel Floor High Exhaust Duct rad monitors and the ~~Refuel Floor~~ Wall Exhaust Duct rad monitors fail to function?

Stem focus?

How do we know radiological consequences? Assumption?

Any release as a result of this accident will:

A. not be processed by SGTS and may result in a site boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

B. not be processed but will be monitored by Zone 1 & Zone 2 rad monitors for conditions requiring entry into EO-100-105, Radioactivity Release Control and an Emergency Plan entry at the Site Area Emergency level.

C. will be processed by Zone 1 and Zone 2 HVAC but still may result in a site boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

D. will be processed by Zone 1 and Zone 2 HVAC but still may result conditions requiring entry into EO-100-105, Radioactivity Release Control and an Emergency Plan entry at the Site Area Emergency level.

---

**Question Data**

**Answer: A** not be processed by SGTS and may result in a site boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

**Explanation/Justification:**

**A.** Correct, inop rad monitors, SGTS won't start, release won't be processed, and release may exceed 10CFR100 limits at site boundary which is 25 Rem whole body and 300 Thyroid.

**B.** Incorrect, Zone 3 independent of 1 and 2, no monitoring done anywhere

**C.** Incorrect, Zone 3 independent of 1 and 2, no impact on site boundary limits by 10CFR 100

**D.** Incorrect, Zone 3 independent of 1 and 2, no processing done anywhere.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295023** | **Refueling Accidents** | **Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS:** | **Entry conditions of emergency plan** |

K/A# **295023.AA2.05** K/A Importance Exam Level **SRO**

References provided to Candidate **None** Technical References: **Tech Spec B3.3.6.2**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

79 A loss of drywell cooling occurs on Unit 1. Drywell pressure increases to 2.9 psig. Drywell temperature increases to 155 degrees F.

Which portions of the Emergency Operating Procedures would the Unit Supervisor be working for the above plant conditions?.

EASY!!!

A. All legs of EO-100-102, RPV Control and all legs of EO-100-103, Primary Containment Control.

B. All legs of EO-100-103, Primary Containment Control. No entry to EO-100-102, RPV Control is required.

C. Primary Containment Pressure (PC/P) and Drywell Temperature (DW/T) legs of EO-100-103, Primary Containment Control. No entry to EO-100-102, RPV Control is required.

D. Primary Containment Pressure (PC/P) and Drywell Temperature (DW/T) legs of EO-100-103, Primary Containment Control and all sections of EO-100-102, RPV Control.

**Question Data**

**Answer: A** All legs of EO-100-102, RPV Control and all legs of EO-100-103, Primary Containment Control.

**Explanation/Justification:**

**A.** correct answer, high drywell pressure would be a scram requiring entry into 102 and 103.
**B.** Must enter 102 for scram.
**C.** Must enter 102 for scram.
**D.** All legs of 103 needed to be worked concurrently.

You are not determining DW temp it is given

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295024** | **High Drywell Pressure** | **Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:** | **Drywell temperature** |

K/A# **295024.EA2.02** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-000-102 or 103**

Question Source: **Modified** **Susquehanna, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

SRO only level? (EOP entry conditions)

K/A mismatch

# SSES LOC 19 NRC Exam

Setpt question
EASY

80 Unit 1 at 100% power when a MSIV isolation occurs causing the following:

- Partial scram with one quarter of the control rods not fully inserted.
- Reactor pressure peaked at 1135 psig.

What automatic actions should have occurred to control the pressure transient and what procedure(s) will be used for reactor vessel pressure control?

typo?

Another way to ask question what is expected plant response at 1135# & for ATWS what EOP entered?

A. Reactor Recirc Pump trip, SRV operation. EO-100-113

B. Main Turbine Bypass valves, grea SRV operation. EO-100-102, EO-100-113.

C. Reactor Recirc Pump trip, Main Turbine Bypass valves. grea EO-100-102

D. ARI, SRV operation. EO-100-102, EO-100-113.

**Question Data**

Reactor Recirc Pump trip, SRV operation. EO-100-113

**Explanation/Justification:**

**A.** correct answer, Recirc pump ATWS trip at 1123, SRV ops at 1126. EO-100-113 for ATWS.

**B.** Bypass valves not available with MSIVs closed. EO-100-102 entered but exited due to more than one rod out, thus not used for pressure control.

**C.** Bypass valves not available with MSIVs closed. EO-100-102 entered but exited due to more than one rod out, thus not used for pressure control.

**D.** EO-100-102 entered but exited due to more than one rod out, thus not used for pressure control.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295025.2.1.23** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **AR-103-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

generic K/A

C
address distractor to address procedure usage

# SSES LOC 19 NRC Exam

81 Unit 1 received a 100% power load reject and a complete loss of offsite power:

- Reactor scram
- 3 SRVs are stuck open
- RPV Pressure 340 psig
- RPV Level -10 inches
- Drywell Pressure +14 psig
- Drywell Temperature 255 F
- Suppression Pool Level 16 feet
- Suppression Chamber Pressure ~~16 psig~~
- Suppression Pool Temperature 270 F

What systems are available for Reactor Vessel Level control?

A. ~~HPCI~~ RCIC, SLC and CRD

B. RHR Service Water, Core Spray, CRD

C. Condensate pumps, RHR, SLC

D. Core Spray, ~~RCIC~~ HPCI, CRD

**Question Data**

**Answer: A** HPCI, SLC and CRD

**Explanation/Justification:**

**A.** correct answer, core spray if flow less than 5200 gpm

**B.** RHR SW doesn't have the required dischargepressure at this time

**C.** No power to the condensate pumps

**D.** HPCI shouldn't be run at >140 deg F due to lube oil cooling. RCIC or HPCI

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295026.2.4.11** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-100-009**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

generic KA

# SSES LOC 19 NRC Exam

82 A LOCA has occurred and level has trended down and the following conditions exist in the plant:

| | |
|---|---|
| Reactor Pressure | 50 psig |
| Wide Range Level | - 150 inches |
| Fuel Zone Level | - 190 inches |
| Upset Range Level | 0 inches |
| Shutdown Range Level | 0 inches |
| Narrow Range Level | 0 inches |
| Drvwell Pressure | + 5.2 psig |
| Drvwell Temperature | 200 degrees F |

As the Unit Supervisor which of the following Reactor Level Instruments would you instruct the operators to use as water level lowers?

A. Shutdown Range — plausible

B. Wide Range

C. Upset Range plausible

D. Fuel Zone Range

Ros should know when RV level instruments are reliable or not.

**Question Data**

**Answer: D** Fuel Zone Range

**Explanation/Justification:**

**A.** 0-500, calibrated cold

**B.** range to -150, calibrated for 1035 reactor pressure

**C.** 0 - 180 calibrated hot.

**D.** correct answer, calibrated for existing conditions, on scale

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295028** | **High Drywell Temperature** | **Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:** | **Reactor water level** |

K/A# **295028.EA2.03** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-145-004**

Question Source: **Modified** **Grand Gulf 1, 2000** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

need better SRO level question

# SSES LOC 19 NRC Exam

83 An ATWS from 100% power has occurred on Unit 1 . The following conditions now exist:

| | | |
|---|---|---|
| RPV level band: | -161 inches to -60 inches | ~~units: is what?~~ |
| Reactor Power: | 25 on IRMs range 6 | |
| SLC Tank level | 5% | |

Based on the above conditions, what will be the next directions given to the shift crew?

A. restore and maintain RPV level in the normal band, +13 to +54 inches

B. commence a controlled cooldown of the reactor vessel per EO-100-102

C. exit the EO-100-113 flowchart and control level per EO-100-102

D. restore and maintain RPV level, -60 to -110 inches

**Question Data**

restore and maintain RPV level in the normal band, +13 to +54 inches

**Explanation/Justification:**

**A.** correct answer, boron injection has occurred for >24 minutes therefore the HSBW has been added and level should be restored to +13-54

**B.** commence a controled cooldown as per EO-100-113, not 102 board.

**C.** remain on the EO-100-113 board for level control.

**D.** with HSBW added, level is restored to +13 to +54 inches.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295031.2.4.22** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-100-113**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

generic KA

# SSES LOC 19 NRC Exam

84 Unit 1 operating at 100% power. 'B' Circ Water (CW) pump motor winding temperature indicates a rising trend on the PICSY computer display and CW pump discharge pressure PI11511A on IC668 indicates a lowering trend.

As the Unit Supervisor which of the following directions to the Plant Control Operators would be appropriate for the given plant conditions.

A. Monitor CW discharge pressure, at 5 psig prior to discharge pressure alarm setpoint - trip CW pump, Monitor Main Condenser vacuum, reduce power per ON-143-001 as required.

B. Monitor CW pump motor temperatures, Monitor Main Condenser vacuum for degradation, monitor Reactor power for a lowering trend, reduce power per GO-100-012 as required.

C. Monitor CW pump motor temperature, at 10 degrees before computer temperature alarm - trip CW pump, monitor Reactor power for a rising trend, reduce power per CRC Book as required.

D. Monitor Main Condenser vacuum for degradation, monitor condensate temperature for a lowering trend, reduce power per ON-143-001 as required.

**Question Data**

**Answer: B** Monitor CW pump motor temperatures, Monitor Main Condenser vacuum for degradation, monitor Reactor power for a lowering trend, reduce power per GO-100-012 as required.

**Explanation/Justification:**

**A.** no discharge pressure alarm on CW system, a trip of the CW pump will cause a recirc runback which is an undesired automatic function

**B.** Correct answer, vacuum lowers due to reduced circ water flow, with reduced vacuum condensate temperature will rise causing a slight reduction in reactor power.

**C.** reactor power will lower not rise due to vacuum and condensate temperature.

**D.** Condensate temperature will rise instead of lower.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295002** | **Loss of Main Condenser Vacuum** | **Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM:** | **Reactor power: Plant-Specific** |

K/A# **295002.AA2.02** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-164-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

85 Given the following conditions on Unit 1:

- Reactor power has been lowered to 95% in preparation for Turbine Control Valve testing
- Prior to starting the test, the PCOM reports APRM reactor power is rising
- Power peaks and stabilizes at 100%
- No alarms are received
- After investigation, the PCOP discovers HPCI is running and injecting
- All other equipment and instruments are operating as designed

HPCI initiation caused by relay room cabinet door jarring relays.

Which of the following is the required Unit Supervisor direction regarding reactor power and the reportablity requirements for these conditions?

The US shall direct a Recirc Flow reduction to:

A. less than 75% power and ensure an 8 hour reportability call is made.

B. less than or equal to 95% power and ~~no~~ 4 hr reportability.

C. less than 75% power and ensure a 4 hour reportability call is made.

D. less than or equal to 95% power and ensure a 4 hour reportability call is made

**Question Data**

**Answer: B** less than or equal to 95% power and ~~no~~ 4 hr reportability.

**Explanation/Justification:**

**A.** power reduction to 75% is required for a loss of feedwater heating.

**B.** correct answer, power is returned to the original starting power and no notification is made since signal was an invalid initiation signal.

**C.** power reduction to 75% is required for a loss of feedwater heating. Incorrect reportability time.

**D.** Incorrect reportability time.

we are NOT determining or interpreting Rx Power

| Sys # | System | Category | K/A Statement |
|---|---|---|---|
| **295014** | **Inadvertent Reactivity Addition** | **Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION:** | **Reactor power** ? |

K/A# **295014.AA2.01** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **NDAP-QA-0720 Attachment G** Technical References: **ON-156-001 & NDAP-QA-0720**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

86 On Unit 1 a scram has occurred. The following plant conditions exist:

- CIG/MSIV interlocks have been bypassed
- Instrument Air supplying CIG
- The SDV is full
- The scram pilot valve air header is "0" psig
- Numerous control rods are NOT fully inserted
- Reactor power is stable at 12%

As the Unit Supervisor reviewing the indications of the failure to scram transient, you direct the PCOM to:

A. Vent the scram air header. plausible

B. Initiate ARI. plausible

C. Reset the scram to drain the SDV and insert a manual scram. obvious

D. De-energize the scram solenoids. plausible

**Question Data**

**Answer: C** Reset the scram to drain the SDV and insert a manual scram.

**Explanation/Justification:**

**A.** Venting the scram air header would not serve any additional purpose since air header is 0.
**B.** ARI would not serve any additional purpose since air header is 0.
**C.** Correct answer
**D.** de-energizing the scram solinoids would not serve any additional purpose since air header is 0.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **295015.2.1.7** K/A Importance **4.4** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-000-113**

Question Source: **Modified** **Duane Arnold, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

generic K/A
ok

rewrite stem and distractors to have something besides "0" psig in air header as a valid distractor

better explanations

# SSES LOC 19 NRC Exam

87 A loss of both CRD pumps has occurred during a reactor startup with reactor pressure at 500 psig.

During the crew brief to discuss the loss of both CRD pumps, an NPO verifies a low pressure accumulator alarm that was received on a rod at position 12? Typo

As the Unit Supervisor which of the following actions are required?

A. Continue the brief, direct the PCOM to check for the second accumulator alarm, then insert manual scram.

B. Suspend the brief, direct the PCOM to insert the rod with the accumulator alarm one notch.

C. Continue the brief, within 20 min. direct the PCOM to restart a pump and insert one rod one notch

D. Suspend the brief, direct the PCOM to immediately insert a manual scram

**Question Data**

**Answer: D** Suspend the brief, direct the PCOM to immediately insert a manual scram

**Explanation/Justification:**

**A.** No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
**B.** No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
**C.** No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
**D.** correct answer, If reactor steam dome pressure < 900 psig and one or more scram accumulators are inoperable, PLACE Reactor Mode Switch in SHUTDOWN position.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **295022** | **Loss of CRD Pumps** | **Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:** | **Accumulator pressure** |

K/A# **295022.AA2.01** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-155-007**

Question Source: **Modified** **Nine Mile Point 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

88 Given the following conditions with Unit 1 in Mode 4 and Unit 2 at 100% power

LOOP/LOCA testing is in progress on Unit 1.
As part of the test, all four Diesel Generators start and load to their respective Unit 1 busses.
ECCS responses are as follows:

| | |
|---|---|
| - 1A RHR Pump | Does NOT start |
| - 1B and 1C RHR Pumps | Start at ~~3~~ 10 seconds |
| - 1D RHR Pump | Starts at 10 seconds |
| - All 4 Unit 1 Core Spray Pumps | Start at ~~10.5~~ 10 seconds |

Which of the following Tech Spec actions should be taken?

A. Declare 1A RHR Pump Inoperable, Unit 1 enters Tech Spec 3.5.2
Declare D DG Inoperable, Unit 1 and 2 enter Tech spec 3.8.2

B. Declare 1A RHR Pump Inoperable, no Tech Spec entry required
Declare D DG Inoperable, Unit 2 enters Tech Spec 3.8.1
D DG can be returned to Operable if 1A RHR Pump breaker DC Knife Switch opened.

C. Declare 1A RHR Pump Inoperable, no Tech Spec entry required
Declare 1D RHR Pump Inoperable, no Tech Spec entry requried
D DG remains Operable

D. Declare 1A and 1D RHR Pumps Inoperable, Unit 1 enters Tech Spec 3.5.2
Declare D DG Inoperable, Unit 2 enters Tech Spec 3.8.1

## Question Data

**Answer: B** Declare 1A RHR Pump Inoperable, no Tech Spec entry required
Declare D DG Inoperable, Unit 2 enters Tech Spec 3.8.1
D DG can be returned to Operable if 1A RHR Pump breaker DC Knife Switch opened.

**Explanation/Justification:**

**A.** 3.5.2 U-1 S/D ECCS is not applicable with one pump out of service. 3.8.2 S/D Electrical is not applicable for unit 2.

**B.** correct answer, With Unit 1 in Mode 4 the RHR pump can be declared out of with the other pump in the loop still operable and no entry into Tech Specs. Surveillance requirements for Unit 2 D/G operability in Mode 1 requires DG auto-starts from standby condition and: energizes permanently connected loads in < 10 seconds, energizes auto-connected shutdown loads through individual load timers. With the RHR pump DC knife switch open the pump not loading does not inop the D/G for unit 2.

**C.** d RHR pump is not inoperable.

**D.** 3.5.2 U-1 S/D ECCS is not applicable with one pump out of service.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **203000** | **RHR/LPCI: Injection Mode (Plant Specific)** | **Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** | **Emergency generator failure** |

K/A# **203000.A2.06** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

89 Given the following conditions:

- Unit 1 is in Mode 4 with Shutdown Cooling in service utilizing the A Residual Heat Removal (RHR) Pump.
- Electrical Maintenance has reported that one of the B RPS MG Set EPA Breakers is running hotter than acceptable
- They recommend B RPS be transferred to the Alternate Power Supply

Which of the following will prevent a loss of Shutdown Cooling (SDC) and addresses the operability issues of opening breakers?

Shutdown Cooling isolation and "A" RHR Pump trip will be prevented by:

needed ?

A. opening the breaker supplying power to SDC Outboard Isolation Valve (HV-151-F008) and taking LCO 3.6.1.3, Condition A.

B. opening the breakers supplying power to SDC Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and taking LCO 3.6.1.3, Condition B.

"taking" an LCO ?

C. opening the breakers supplying power to SDC Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and taking LCO 3.3.6.1, Condition B.

D. opening the breaker supplying power to SDC Outboard Isolation Valve (HV-151-F008) and NO LCO is required due to the plant being in Mode 4.

---

**Question Data**

**Answer: B** opening the breakers supplying power to SDC Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and taking LCO 3.6.1.3, Condition B.

**Explanation/Justification:**

**A.** Incorrect, loss of B RPS cause both F008 and F009 to close.
**B.** Correct, allowed by procedure
**C.** Incorrect, not an instrumentation LCO
**D.** ncorrect, loss of B RPS cases both F008 and F009 to close, must take LCO

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **205000** | **Shutdown Cooling System (RHR Shutdown Cooling Mode)** | **Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM/MODE including:** | **Pump trips** |

K/A# **205000.A3.02** K/A Importance ~~**3.2**~~ ? Exam Level **SRO**

References provided to Candidate **TS 3.3.6 & 3.6.1** Technical References: **OP-149-002**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

all TS Sect 3 will be provided

# SSES LOC 19 NRC Exam

90 Given the following conditions with Unit 1 at 100% power:

- Core Spray Loop A Header Break Detection High Differential Pressure alarm has just been received
- The Reactor Building NPO reports PDIS-E21-1N004A is reading -3.4 psid on Panel 1C010.

alarm normal indication

Which of the following actions are required for these conditions?

How do we know the instrument is inop?

∴ Grant inop

A. Declare Core Spray A header d/p instrumentation channel inoperable. Restore to Operable status within 72 hours.

B. Declare Core Spray Loop A Inoperable. Restore to Operable status within 72 ~~days~~ hours.

C. Write an AR to document the out-of-specification differential pressure condition.

D. Write a tracking LCO to document the system's ability to inject inside the vessel but not spray above the core.

---

**Question Data**

**Answer: A** Declare Core Spray A header d/p instrumentation channel inoperable. Restore to Operable status within 72 hours.

**Explanation/Justification:**

**A.** Correct.
**B.** Incorrect. CS still operable, d/p still good, bad alarm
**C.** Incorrect. D/P still good, AR should be for incorrect alarm received
**D.** Incorrect. CS still capable of performing its intended function

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **209001** | **Low Pressure Core Spray System** | **Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** | **Core spray line break** |

K/A# **209001.A2.05** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **TRO 3.5.2** Technical References: **TRO 3.5.2, Pages 3.5-3 through 3.5-5**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

all inst. TRO's provided

# SSES LOC 19 NRC Exam

91 Given the following conditions with Unit 2 in Mode 1:

- Surveillance SO-024-001, Monthly Diesel Generator Operability Test, is being performed for the A DG to the 2A201 bus.
- The 2A201 bus is currently loaded at 800 KW
- The A DG is operating at the SO hold point of 1000 KW.
- The Supply Breaker to 2A201 from Transformer 0X201 trips due to a breaker problem.

Which of the following describes the expected electric plant response and the Unit Supervisor directed actions for these conditions?

The A Diesel Generator Output Breaker:

A. trips and the US should direct resetting the 2A201 bus lockouts and power restoration from its alternate power source.

B. trips and the US should direct verification of 2A201 automatically re-energizing from its alternate power source.

C. does NOT trip and the US should direct restoration of normal bus voltage and frequency parameters.

D. does NOT trip and the US should direct an immediate trip of the Diesel Generator.

---

**Question Data**

**Answer: C** does NOT trip and the US should direct restoration of normal bus voltage and frequency parameters.

**Explanation/Justification:**

**A.** Incorrect, breaker does not trip, no lockouts trip, bus remains energized
**B.** Incorrect, breaker does not trip, bus remains energized
**C.** Correct, a loss of 200 KW will show up as changes in frequency and voltage, return parameters to normal bands.
**D.** Incorrect, DG can handle this transient without breaker or engine tripping, no reason to direct trip

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **264000** | **Emergency Generators (Diesel/Jet)** | **Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** | **Loss of A.C. power** |

K/A# **264000.A2.09** K/A Importance **4.1** Exam Level **SRO**

References provided to Candidate **None** Technical References: **Electrical Theory and application**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

need proc for control of EDG parameters

# SSES LOC 19 NRC Exam

92 Unit 1 is at rated conditions with no LCOs entered.

Electrical Maintenance has submitted a work package to overhaul the HV-151-F006A Shutdown Cooling Suction Valve motor actuator.

The work package requires the HV-151-F004A to be closed to allow manual operation of the HV-151-F006A to set the valve limit switches .

What Limiting Conditions of Operation will be in effect?

(changes in answers made by licensee prior to validation)

A. ~~LCO 3.4.8, LCO 3.4.9, LCO 3.6.1.3~~ No LCO reqd.

B. LCO 3.4.8, LCO 3.4.9, LCO 3.5.1, LCO 3.6.1.3

C. ~~[illegible]~~ LCO 3.5.1 STE ~~no LCO reqd~~

D. ~~LCO 3.4.9~~ 3.5.1, LCO 3.6.2.3, LCO 3.6.2.4

**Question Data**

**Answer: C** LCO 3.5.1

**Explanation/Justification:**

**A.** SDC not required at rated conditions.
**B.** SDC not required at rated conditions.
**C.** correct answer, to open the 06, the 04 suction valve must be closed, taking 'A' RHR pump out of service.
**D.** ECCS LCO required for RHR pp out of service.

| Sys # | System | Category | KA Statement |
|---|---|---|---|

K/A# **219000.2.2.24** K/A Importance **3.8** Exam Level **SRO**

References provided to Candidate **TS 3.4, 3.5 & 3.6** Technical References:

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

generic K/A for RHR/LPCI dk

# SSES LOC 19 NRC Exam

93 With the plant in Mode 5 and the Refueling Platform approaching the reactor vessel, which of the following will initiate a rod withdraw block for any selected rod?

A. Grapple NOT engaged.

B. Any Hoist extended.

C. Any rod NOT fully inserted.

D. Any Hoist Loaded.

---

**Question Data**

**Answer: D** Any Hoist Loaded.

**Explanation/Justification:**

**A.** grapple not engaged will not cause a rod block.
**B.** no interlock for hoist extended, everything uses load cells.
**C.** 1 rod allowed full out.
**D.** Correct answer,

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| **234000** | **Fuel Handling Equipment** | **Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including:** | **Interlock operation** |

K/A# **234000.A3.02** K/A Importance **3.7** Exam Level **SRO**

References provided to Candidate **None** Technical References: **TM-OP-56**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

94 Unit 2 is operating at 100% power when fuel damage occurs, the manual scram fails. will for stem focus

The following plant conditions exist:

| | |
|---|---|
| - Reactor power | 18% |
| - Reactor pressure | 940 psig |
| - RPV water level | -100 inches |
| - Main Steam Line B | Inboard & Outboard MSIVs failed open |
| - Main Turbine | Tripped |
| - Site boundry release (Adult Thyroid) | 4.90 Rem(rising) |

Given the above conditions, which of the following actions are required?

A. Use the SRVs to maintain reactor pressure less than 965 psig.

B. Use Main Turbine BPVs to commence a reactor cool down at less than a 90F/Hr rate.

C. Use Main Turbine BPVs to commence a reactor cool down at greater than a 90F/Hr rate.

D. Perform Emergency RPV Depressurization.

---

**Question Data**

**Answer: D** Perform Emergency RPV Depressurization.

**Explanation/Justification:**

**A.** EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

**B.** EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

**C.** EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

**D.** correct answer, MSIVs failing to close is a "Primary system" discharging. The ATWS is being controlled by EO-200-113 with level being maintained at -100" as required by the procedure.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of refueling administrative requirements.** |

K/A# 2.2.26 typo K/A Importance 4.2 Exam Level SRO

References provided to Candidate **None** Technical References: **EOP Flow charts**

Question Source: **Modified** **Fermi 2 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

K/A 2.1.20 Ability to execute procedural steps (ie EOP flowchart step determination)

# SSES LOC 19 NRC Exam

95 In accordance with Unit 1 Tech Specs and ON-183-001, Stuck Open Safety Relief Valve, the Reactor Mode Switch was placed in Shutdown at 108 degrees F due to a stuck open SRV.

Post-scram, the Suppression Pool reached a peak of 114 degrees F before Suppression Pool Cooling was able to begin removing heat. The reactor was NOT required to be placed in Mode 4.

Which of the following describes the restrictions on the ensuing reactor startup? Assume the SRV has been repaired.

Suppression Pool temperature must be less than or equal to:

A. 105 degrees F prior to placing the Reactor Mode Switch in Startup/Standby.

B. 90 degrees F prior to placing the Reactor Mode Switch in Startup/Standby.

C. 90 degees F prior to exceeding 1% power.

D. 105 degrees F prior to exceeding 1% power.

---

**Question Data**

**Answer: C** 90 degees F prior to exceeding 1% power.

**Explanation/Justification:**

**A.** Incorrect, 105 limit only applies >1% power with testing in progress

**B.** Incorrect, no requirement for this since the reactor remains in one of the three modes for which the LCOs apply at all times

**C.** Correct, in Modes 1, 2 and 3 this limit applies once >1% power.

**D.** Incorrect, 105 limit only applies >1% power with testing in progress.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Conduct of Operations** | **Knowledge of less than one hour technical specification action statements for systems.** |

K/A# **2.1.11** K/A Importance ~~**3.8**~~ Exam Level **SRO**

References provided to Candidate **TS 3.6.2** Technical References: **TS 3.6.2.1**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

Good all TS Sect 3 without Bases

# SSES LOC 19 NRC Exam

96 Which of the following is/are considered an Unreviewed Safety Question?

1. Emergency actions that depart from T.S. are needed to protect the public health and safety.
2. The possibility of an accident exists that has not been evaluated by the FSAR
3. The consequence of a malfunction of equipment evaluated by the FSAR is increased.
4. The margin of safety as defined in T.S. is reduced.
5. An emergency event that can not be classified by the Emergency plan.

A. 2, 3, 4

B. 1, 2, 3

C. 3, 4, 5

D. 2, 3, 5

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**Question Data**

**Answer: A** 2, 3, 4

**Explanation/Justification:**

**A.** correct answer, Not USQ - Emergency actions that depart from T.S. are needed to protect the public health and safety.
Is USQ - The possibility of an accident exists that has not been evaluated by the FSAR
Is USQ - The consequence of a malfunction of equipment evaluated by the FSAR is increased.
Is USQ - The margin of safety as defined in T.S. is reduced.
Not USQ - An emergency event that can not be classified by the Emergency plan.

**B.** 1, not USQ

**C.** 5, not USQ

**D.** 5, not USQ

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of the process for determining if the proposed change, test, or experiment involves an unreviewed safety question.** |

K/A# **2.2.8** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References:

Question Source: **Modified** **Columbia Gen Sta 2, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

97 Which of the following is the bases for the Technical Specifications, Minimum Suppression Chamber Water Volume in Operation Conditions 1, 2, and 3?

A. Ensures a sufficient supply of water is available, with the Minimum CST Volume in the event of a LOCA to permit recirculation cooling flow to the core.

B. Ensures a sufficient amount of water would be available to adequately condense the steam from the SRV tailpipe break above the suppression pool level.

C. Ensures a sufficient amount of water would be available to adequately condense the steam from the SRV discharges, downcomers, or HPCI and RCIC turbine exhaust lines and provide emergency make-up to the reactor vessel.

D. Provides sufficient supply of water that, with the Minimum CST Volume, Long Term Cooling is available for the design basis accident.

---

**Question Data**

**Answer: C** Ensures a sufficient amount of water would be available to adequately condense the steam from the SRV discharges, downcomers, or HPCI and RCIC turbine exhaust lines and provide emergency make-up to the reactor vessel.

**Explanation/Justification:**

**A.** CST is not part of basis.

**B.** HPCI & RCIC are included in the basis.

**C.** correct answer, If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the SRV discharges, downcomers, or HPCI and RCIC turbine exhaust lines##Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System##The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

**D.** CST is not part of basis.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Equipment Control** | **Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.** |

K/A# **2.2.25** K/A Importance **3.7** Exam Level **SRO**

References provided to Candidate **None** Technical References: **TS Basis - B 3.6.2.2**

Question Source: **Modified** **Fermi 2 2, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

98 Maintenance must be performed in the Unit 1 Reactor Water Cleanup (RWCU) Backwash Receiving Tank Room (Room 1-509) to support a modification.

Per NDAP-QA-0323, Standard Blocking Practices, and NDAP-QA-0626, Radiologically Controlled Area Access and Radiation Work permit (RWP) System, which of the following actions is required BEFORE the Unit Supervisor may allow work to begin in the room to comply with the ALARA BLOCKING principle?

A. Flush, then drain the Backwash Receiving Tank and maintain the tank empty.

B. Backwash the RWCU filters, then drain the filters and maintain the filters empty.

C. Flush and drain, then fill the Backwash Receiving Tank and maintain the tank full.

D. Backwash the RWCU filters, then fill the tank and maintain the tank full.

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**Question Data**

**Answer: C** Flush and drain, then fill the Backwash Receiving Tank and maintain the tank full.

**Explanation/Justification:**

**A.** The tanks must also be filled with water to act as shielding.

**B.** This would not help the radiation levels in the Backwash Receiving Tank Rooms and may make it worse since the filters are backwashed to the room.

**C.** correct answer, NDAP-QA-0323, Standard Blocking Practices requires the tanks be flushed drained and filled before entry into the rooms.

**D.** This would not help the radiation levels in the Backwash Receiving Tank Rooms and may make it worse since the filters are backwashed to the room.

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| | | | |
|---|---|---|---|
| Sys # | System | Category | KA Statement Implement Procedures to Reduce Rad Exposure |
| K/A# 2.3.10. | K/A Importance 3.3 | Exam Level SRO | |
| References provided to Candidate | None | Technical References: | NDAP-QA-0626, Sect. 6.2.5 |
| Question Source: | Modified Susquehanna, 2001 | Level Of Difficulty: (1-5) | 3 |
| Question Cognitive Level: | Memory | 10 CFR Part 55 Content: | 55.43 |

# SSES LOC 19 NRC Exam

99 Maintenance has just reported to the Control Room that a 55 gallon drum of lube oil leaked into the River Water Makeup intake bay overnight.

Using the attachments from NDAP-QA-0720, STATION REPORT MATRIX AND REPORTABILITY EVALUATION GUIDANCE, determine ALL of the offsite agencies that must be notified after PA DEP is notified.

A. LCEMA, Coast Guard, NRC

B. CCEMA, Coast Guard, NRC

C. LCEMA, Coast Guard, PEMA

D. LCEMA, CCEMA, PEMA

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**Question Data**

**Answer: A** LCEMA, Coast Guard, NRC

**Explanation/Justification:**

**A.** correct answer, PA DEP, LCEMA, and Coast Guard must be notified due to a petroleum product being discharged to a waterway. The NRC must be notified anytime an offsite agency is notified.

**B.** NRC must be notified.

**C.** Correct for a Comprehensive Environmental Response, Compensation, and Liability Act release which this spill is NOT.

**D.** Columbia county not notified for this spill not PA Emergency Management Agency.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Emergency Procedures and Plan** | **Knowledge of which events related to system operations/status should be reported to outside agencies.** |

K/A# **2.4.30** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **NDAP-QA-0720 Att Q & T** Technical References: **NDAP-QA-0720**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

# SSES LOC 19 NRC Exam

100 The Mode Switch has been placed in Shutdown.
Reactor Power is 15%.
Reactor Water Level is lowering at 1 inch per minute.
Drywell pressure is 0.75 psig and slowly rising.

The Unit Supervisor is about to give direction from EO-100-113, LEVEL/POWER CONTROL to inhibit ADS and bypass MSIV and CIG interlocks, why?

A. Current level control range is near the auto initiation of ADS which could allow uncontrolled low pressure injection and main condenser is used as a heat sink for as long as possible. syntax

B. Initiation of ADS and/or closure of the MSIVs would cause large pressure changes. GE calculations indicate swings in pressure cause large power oscillations.

C. Initiation of ADS would cause large pressure changes with resulting power oscillations. MSIVs must remain open for use of the main condenser as a heat sink.

D. MSIVs must remain open for use of the main condenser as a heat sink. CIG must remain available for later use of ADS valves.

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**Question Data**

**Answer: A** Current level control range is near the auto initiation of ADS which could allow uncontrolled low pressure injection and main condenser is used as a heat sink for as long as possible.

**Explanation/Justification:**

**A.** correct answer, Prevent low pressure unborated injection and prevent MSIVs from clossing due to a loss of pneumatic supply.
**B.** ADS would depressurize the RPV.
**C.** ADS would depressurize the RPV.
**D.** CIG not required, 2200# bottles supply header.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| | | **Emergency Procedures and Plan** | **Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.** |

K/A# **2.4.49** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **None** Technical References: **EO-000-113**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**