QUESTION 1

- The plant was at 50% power with 4 Recirc Pumps in operation, following a short forced outage, and restoration of rated power is underway. A malfunction occurred in the master recirculation controller which caused recirculation flow and reactor power to lower.
-

The Reactor Operator has taken all recirculation speed
- controllers to MANUAL and the flow/power reduction has ceased. The
- following conditions exist:

  - Reactor power is 30% and steady
  - Reactor recirculation flow is 20% Core Rated Flow

Which of the following actions are required?

A. Manually Scram the Reactor.

B. Raise recirculation flow or insert rods.

C. Raise recirculation flow by starting the idle recirc Pump.

D. Perform a normal plant shutdown per N1-OP-43C.

K&A #	295001 K1.02
Importance Rating	3.3

QUESTION 1

K&A Statement: Knowledge of the operational implications of Power/Flow distribution as it applies to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION.

Justification:

A. Incorrect but plausible if the operator uses the 5 loop power flow map, then plant is in manual scram zone.

B. Correct – plant is in restricted zone. Per N1-SOP-1.5 exit restricted zone by raising recirc flow or lower power by inserting cram rods.

C. Incorrect but plausible, however 5.2 of N1-SSOP-1.5 states starting a recirc pump to increase recirc flow while operating in the restricted zone is prohibited.

D. Incorrect because N1-SOP-1.5 requires the restricted zone to be exited and directs actions to either raise recirc flow or lower power by inserting the cram rods to 00. This answer is plausible because it would have the operator exit the restricted zone however it is not correct because it is not required to shutdown the reactor.

References:	N1-SOP-1.5, Power Flow Map 4 and 5 loop	Student Ref:	Power Flow Map 4 and 5 Loop. Zones Not Labeled.

Learning Objective:	N/A
Question source:	Modified OC 2006 NRC
Question History:	None
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis: X
10CFR	41.8
Comments:	

QUESTION 2

Plant is operating at 50% power with the following equipment tagged out of service:

- #14 Reactor RP Motor MG Set
- #11 Condenser Circulating Water Pump
- #12 Condensate Pump
- #12 Service Water Pump

- PB 12 has a fault and supply breakers R122 and R123 are tripped.

What is the correct operator response and what is the reason for that response?

A. Scram the Reactor because only 2 Reactor Recirc Pumps are running.

B. Scram the Reactor because Condenser Circulating Water Pumps are not available.

C. Scram the Reactor because only 3 Reactor Recirc Pumps are running.

D. Start Emergency Service Water Pumps and Scram the Reactor because Service Water pumps are not available.

K&A # 295003 K3.05
Importance Rating RO-3.7

QUESTION 2

K&A Statement: Knowledge of the reasons for the following responses as they apply to Partial or Complete Loss of AC: Reactor Scram

Justification:

A. Incorrect- Plausible if the candidate does not know that #11, 12 and 13 will still be available with PB 12 de-energized.

B. Correct – #11 Condenser Circulating Water Pump is tagged OOS and the loss of PB 12 which will cause a loss of #12 Condenser Circulating Water Pump. N1-SOP-30.2 Requires a reactor scram if no circ water pumps are running.

C. Incorrect. 3 Recirc pumps are operating, however the plant is allowed to operate with only 3 loops. Plausible if the applicant does not realize this.

D. Incorrect. Plausible if the applicant does not realize Service Water Pump 11 is available. N1-SOP-18.1 requires scram if neither service water pump can be started.

References: N!-OP-30, N1-SOP-30.2, N1-SOP-18.1 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5

QUESTION 3

During a Station Blackout, which one of the following is the reason for performing designated Battery Load reductions at the start of the station blackout?

A. Maintain power to reactor instrumentation, EC controls, and to start an EDG.

B. Maintain power to the Process Computer and Annunciators.

C. Avoid a loss of critical battery board loads due to breaker trips on over current.

D. Maintain power to emergency DC bearing oil pump and Control Room Emergency lighting.

K&A # 295004 K2.01
Importance Rating 3.1

QUESTION 3

K&A Statement: Knowledge of the reasons for the following responses as they apply to Partial or Complete Loss of DC Power: Load Shedding.

Justification:

A. Correct - SBC's are lost; preserving DC power for Rx Instrumentation, EC controls, and starting EDGs is critical in a blackout.

B. Incorrect. Plausible if the applicant does not know what loads are shed. PPC and annunciators are de-energized when MG 167 is tripped.

C. Incorrrect. Plausible if the applicant does not know that the breakers on Safety Related batteries are bypassed by fuses.

D. Incorrect-Plausible if the applicant does not know these loads are on the Non Safety Battery 14 and the loads are not shed from this battery.

References: N1-SOP-33A.2, N1-OP-47A Student Ref: None

Learning Objective: N/A

Question source: NMP1 Modified.

Question History: 2004 NRC Test

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 4

The reactor is at 42% power and the turbine generator has just been
• placed on-line. The Reactor Operator is raising reactor power by withdrawing control rods.

Which one of the following is correct?

A. A loss of both stator cooling pumps requires an immediate turbine trip ONLY per N1-OP-44.

B. A turbine Journal vibration of 15 mils (steady) requires an immediate reactor scram and turbine trip per N1-OP-31.

C. A loss of both stator cooling pumps requires an immediate reactor scram and turbine trip per N1-OP-44.

D. A turbine Journal vibration of 15 mils (steady) requires an immediate turbine trip ONLY per N1-OP-31.

K&A # 295005 G2.1.32
Importance Rating 3.8

QUESTION 4

K&A Statement: Ability to explain and apply system limits and precautions as they relate to MAIN TURBINE GENERATOR TRIP.

Justification:

A. Incorrect but plausible, however if a governor runback occurs due to Stator Water Cooling System problems and inlet water conductivity was less than 0.5 micromhos/cm the generator may carry remaining load for a period of one hour prior to being tripped.

B. Incorrect but plausible however the turbine can be tripped <45% power without removing the reactor from service.

C. Incorrect but plausible, however if a governor runback occurs due to Stator Water Cooling System problems and inlet water conductivity was less than 0.5 micromhos/cm the generator may carry remaining load for a period of one hour prior to being tripped.

D. Correct – N1-OP-31precaution and limitation 6.0 gives guidance that >12 mils trip the turbine immediately, and section H.7.0 gives guidance for tripping the turbine <45% power.

References: N1-OP-31, N1-OP-44 Student Ref: None

Learning Objective: N/A

Question source: Modified OC 2006 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 5

Initial plant conditions are as follows:

A plant startup is in progress with

- reactor power at 10%
    - The mode switch is in STARTUP
    - Recirculation flow is 50% of rated core flow.
    - Reactor pressure is 1000 psig

A turbine bypass valve malfunction causes:

- A spike in reactor pressure to 1063 psig
- A spike in reactor power to 40%

What is the status of the reactor?

A. At power

B. Scrammed due to high reactor pressure

C. Scrammed due to high IRM neutron flux

D. Scrammed due to high APRM neutron flux

K&A # 295006 K2.06
Importance Rating 4.2

QUESTION 5

K&A Statement: Knowledge of the interrelations between SCRAM and the following: REACTOR POWER.

Justification:

A. Incorrect but plausible if the applicant does not realize that a IRM range 10 scram setpoint has been exceeded.

B. Incorrect but plausible if the applicant does not know that the setpoint for high reactor setpoint is 1080psig.

C. Correct - based on the conditions given (STARTUP, at 10% power), the reactor is operating on IRM Range 10. The scram setpoint for IRM Range 10 is 38.4% (LSSS), which was exceeded.

D. Incorrect but plausible if the applicant does not know that with 50% rated flow the APRM flow biased scram setpoint would be in excess of 90% (.5w+ 67%) which is much higher than the IRM setpoint of 38.4%.

References: Tech Spec 2.1.2.b, Power Flow Operating Map

Student Ref: None

Learning Objective: N/A

Question source: OC 2006 NRC Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 6

The Control Room must be abandoned due to a fire. The immediate actions of N1-SOP-21.2 have been taken and an operator is standing by Remote Shutdown Panel 11.

In order to disable the control room controls and actuate the local panels the operators must place the Channel 11 Control Transfer keylock switch to the _______(1)_______ position. These actions ______(2)_____ Emergency Condenser Isolation.

A. (1) LOCAL
(2) Will prevent

B. (1) EMERG
(2) Will **NOT** prevent

C. (1) LOCAL
(2) Will **NOT** prevent

D. (1) EMERG
(2) Will prevent

K&A # 295016 A1.07
Importance Rating 4.2

QUESTION 6

K&A Statement: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Control room/local control transfer mechanisms.

Justification:

A. Incorrect but plausible, The EC Level control transfer switch position is local. Also the isolation is not prevented.

B. Correct – The transfer switch position is EMERG. If the EC isolates, then the EC Isolation Bypass switch must be placed in bypass.

C. Incorrect but plausible, the EC Level control transfer switch position is local. Not the CH 11 control transfer keylock switch.

D. Incorrect but plausible, the EMERG switch position is correct however it does not prevent auto isolation.

References: N1-OP-13, N1-SOP-21.2 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 7

- The Plant is operating at 90% power with one Reactor Building Closed Loop Cooling (RBCLC) pump tagged out of service.
- An electrical problem causes one of the running RBCLC pumps to trip.
-

H1-4-1 (R Building Cooling Water Press-Temp Makeup Flow) is in alarm.

-
- N1-SOP-11.1 RBCLC Failure has been entered.

Which of the following components should be removed from service to reduce the heat load on the RBCLC system?

A. Reactor Water Cleanup Pumps.

B. Drywell Air Coolers.

C. Fuel Pool Heat Exchangers.

D. Air compressors inter and after coolers.

K&A # 295018 K1.01
Importance Rating 3.5

QUESTION 7

K&A Statement: Knowledge of the operational implications on the effects on component/system operations as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER.

Justification:

A. Correct –Per N1-SOP-11.1 directs that if RBCLC cooling is challenged then trip RWCU pumps. 5.2 states removal of RWCU can quickly remove a large heat load on the system with minimal plant impact.

B. Incorrect but plausible, Drywell air coolers are a major heat load, however equipment is only to be removed from service if a high temp. condition exists per N1-SOP-11.1.

C. Incorrect but plausible, the fuel pool heat exchangers are a minor heat load and removing them from service will not correct the degraded situation.

D. Incorrect but plausible, the air compressor inter and after coolers are minor loads and running the air compressors with no cooling will cause them to trip having a large impact on operation of the plant.

References: N1-SOP-11.1 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.8-
41.10

QUESTION 8

The Plant is operating at 94% power when an electrical disturbance occurs resulting in the following:

- R-10 is OPEN
- R-40 is OPEN
- Reactor has scrammed
- Diesel Generator 103 has failed to start

Considering the above conditions, which one of the following air compressors can be lined up to supply instrument air?

A. Instrument Air Compressor 13

B. Instrument Air Compressor 11

C. Service Air Compressor

D. Instrument Air Compressor 12

K&A # 295019 A1.03
Importance Rating 3.0

QUESTION 8

K&A Statement: Ability to operate and/or monitor the Instrument Air compressor power supplies as they apply to Partial or Complete Loss of Instrument Air.

Justification:

A. Incorrect but plausible if the applicant does not realize that R-10 and R-40 being open means that non safety related air compressors do not have power.

B. Correct – EDG 102 has started and can supply IAC 11.

C. Incorrect but plausible if the applicant does not realize that R-10 and R-40 being open means that non safety related air compressors do not have power.

D. Incorrect but plausible if the applicant does not realize that IAC 12 can only be powered from EDG 103 which failed to start.

References: SOP-20, C-19436-C, C19439-C Student Ref: None

Learning Objective: N/A

Question source: NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 9

The plant was shutting down for an outage, with the following conditions:

- RPV coolant temperature is 195 F and is cooling down.
- Shutdown Cooling pumps 11 and 12 are in service with a total SDC system flow of 4000 GPM.
- RBCCW flow through each of the in-service SDC heat exchangers is 1000 GPM.
- 3 Reactor Recirculation pumps are running.
- 2 Reactor Recirc pumps are tagged out for maintenance.
- RPV water level is being maintained at 73 inches indicated level.

- All of the Reactor Recirculation pumps trip. The cause has not been determined.

The new plant conditions are as follows:

- Shutdown Cooling flow has been verified at 4000 GPM

Which of the following actions will prevent thermal stratification in the reactor?

A. Maximize SDC flow with SDC pumps 11 and 12 per N1-OP-4.

B. Raise reactor level above the main steam line nozzles.

C. Maximize SDC flow by placing in service SDC pump 13 per N1-OP-4.

D. Raise reactor level above the main steam line nozzles and shut all recirc pump suction valves.

K&A # 295021 A2.02
Importance Rating 3.4

QUESTION 9

K&A Statement: Ability to determine and/or interpret RHR/shutdown cooling system flow as they apply to LOSS OF SHUTDOWN COOLING.

Justification:

A. Incorrect but plausible, but just increasing flow will not ensure that SDC cooling flow will go through the core because there is still a bypass path through the recirc valves.

B. Incorrect but plausible if the applicant does not realize that by leaving the recirc valves open, the SDC flow will bypass the reactor.

C. Incorrect but plausible, 3 SDC pump operation needs vessel level to be at the reactor flange for NPSH requirements.

D. Correct – Per N1-OP-4 with no recirc pumps running, reactor level needs to be maintained above the main steam line nozzles with the SDC system in service and all recirc pump suction valves closed or discharge and discharge bypass valves closed.

References: N1-OP-4 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 10

The plant is in a refueling outage with the following:

- A fuel bundle, being moved from its core location to the Spent Fuel Pool, has just been dropped from the grapple.
- The dropped bundle is now in the transfer canal.
- The Refueling Bridge High Range Radiation Alarm is alarming.

Which one of the following identifies the actions required by N1-SOP-34, Dropped Fuel Assembly?

A. Direct personnel remaining in the area to stand clear of the canal.

B. Verify the Control Room Emergency ventilation system is in standby.

C. Verify the Reactor Building Emergency Ventilation system has initiated.

D. Notify the Shift Manager to initiate a protected area evacuation.

K&A # 295023 A1.07
Importance Rating 3.6

QUESTION 10

K&A Statement: Ability to operate and/or monitor the Standby Gas Treatment System as they apply to REFUELING ACCIDENTS.

Justification:

A. Incorrect but plausible, the actions are the correct actions for a new fuel bundle per N1-SOP-34, however the bundle is an irradiated fuel bundle. Dropping a spent fuel bundle requires the refuel floor and drywell to be evacuated.

B. Incorrect but plausible if the applicant does not know that the CREVS System initiates on the Fuel Pool High Alarm per N1-OP-50B.

C. Correct – Per N1-OP-50B section 8.0 the RB emergency ventilation system will initiate, and N1-SOP-34 requires to verify the system has started.

D. Incorrect but plausible if the applicant does not realize that the fuel floor and drywell need to be evacuated vice a protected area evacuation.

References: N1-SOP-34, N1-OP-50B Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 Bank

Question History: NRC 2006 #58

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 11

Following a small LOCA in the drywell the following events occur at the noted times:

02:18 - Venting primary containment through RBEVS initiated
02:47 - The reactor is manually scrammed
02:51 - Drywell pressure spikes to 5.0 psig and then dropped to 3.0 psig and is stable
03:03 - RPV water level dropped to +15 inches and is now stable
03:23 – Condenser Vacuum is at 17 in Hg and dropping

Based on the above conditions, which one of the following is isolated and requires the isolation signals to be bypassed in order to place the system in service?

Reactor Coolant Sampling
Containment H2-O2 Monitoring
Reactor Water Cleanup

A. None

B. Reactor Coolant Sample Valves

C. Containment H2-O2 Monitor capability

D. Reactor Water Cleanup

K&A #	295024 K2.07
Importance Rating	3.9 (RO) / 4.0 (SRO)

QUESTION 11

K&A Statement: **K2.05** -Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: PCIS/NSSSS

Justification:

A. A is incorrect – The containment isolation signals are required to be bypassed to re-establish H2/O2 monitoring capability. This is a valid distractor for those candidates that do not recognize the sealed in containment isolation signal present after drywell pressure reached 5.0 psig.

B. B is incorrect – Sample valves are still open. They isolate on a vessel isolation which occurs at +5 inches or 7 in. Hg, and do not isolate on containment isolation. This is a plausible distractor because in the EOP-1 Attachment 10 has steps to bypass the auto vessel or containment isolations if they have occurred. Since the vessel isolation has not occurred the isolation signal is not required to be bypassed to obtain vessel samples.

C. CORRECT C is correct - Containment isolation has occurred with drywell pressure reaching 5.0 psig. The return of drywell pressure to 3.0 psig does not auto reset after dropping below 3.5 psig. H2-O2 monitoring is bypassed and restored per EOP-1 Attachment 11 and requires the CAD Channel 11 and 12 bypass switches to be placed in bypass prior to opening the Containment Monitoring System Isolation valves.

D. D is incorrect – Cleanup isolates on a vessel isolation and not containment isolation. This action is taken following cleanup isolation to restore cleanup to lower reactor water level, not to raise it. This is a valid distractor for those candidates who believe RWCU isolates on containment isolation signals other that 5" Rx level.

References: N1-SOP-40.2 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2005 NRC Exam

Question History: NMP1 2005 NRC Exam

Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X

10CFR 55 41.7

QUESTION 12

The plant was operating at 100% power when a spike in reactor pressure caused a reactor scram due to high flux.

Which one of the following could be a cause of the pressure spike and scram?

A. During the previous Turbine Startup the Load Limit was not moved fully out of the way after the Speed Governor was in control of Turbine Speed.

B. During the previous Reactor Startup the MPR to EPR controlling setpoint difference was set at 10 psid.

C. The MPR failed to the high end of the pressure setpoint for the MPR.

D. The EPR failed to the low end the pressure setpoint for the EPR.

K&A # 295025 K2.08
Importance Rating 3.7

QUESTION 12

K&A Statement: Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/Turbine pressure regulating system: Plant specific.

Justification:

A. Incorrect. The Load limit would not cause a spike. If it was not moved out of the way it could limit the ability to raise load. Plausible because it is one of the turbine controls that will affect pressure and load.

B. Correct – The MPR to EPR setpoint should be within 5 psid. IF the EPR fails, the MPR will take control and the pressure spike if it too large can cause a scram on high flux.

C. Incorrect. The MPR failing to the high pressure setpoint would try to cause the Turbine Control valves to close to raise pressure. However the EPR would maintain control in this situation and the MPR would have no affect. Plausible if the applicant does not realize this relationship between the MPR and EPR.

D. Incorrect. The EPR failing to a low pressure setpoint would attempt to open the control valves further. This would cause reactor pressure to go down not spike up. Plausible if the applicant does not realize the EPR will open the control valves in this situation.

References: N1-OP-31 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

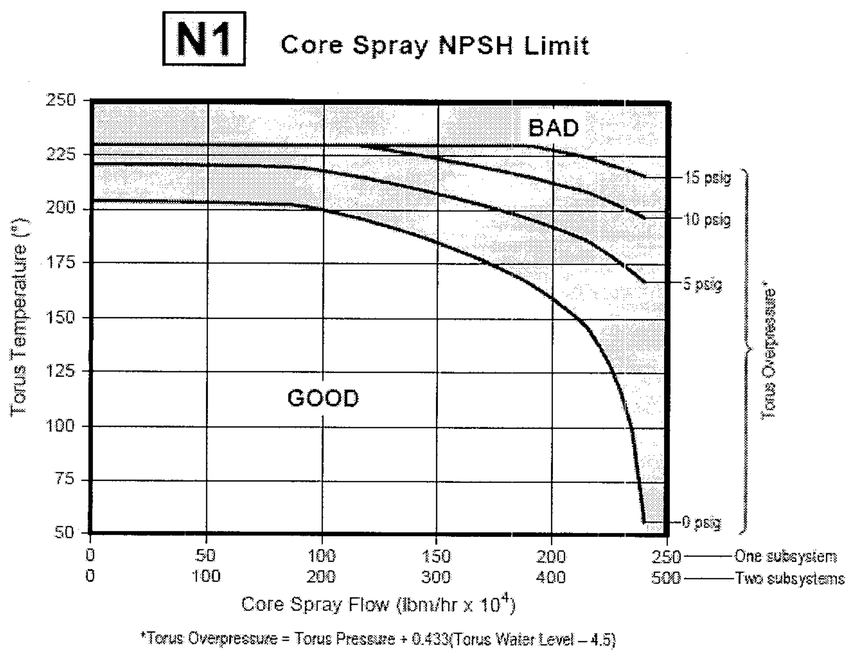
10CFR 41.7

QUESTION 13

A loss of coolant accident has occurred, with the following:

- The RPV has been depressurized using EC's and ERV's
- Core Spray is injecting and maintaining reactor level
- Containment Sprays have been utilized to lower Torus pressure
- Torus Water temperature is 200°F
- Torus Level is 9.5 feet
- Torus pressure is 6 psig
- Drywell pressure 7 psig
-
- Electrical faults in the plant have resulted in valves 40-01 and 40-12 going closed.

Which one of the following states the maximum Core Spray flow (lbm/hr x 10E4) that may be used for RPV injection?



A. 240

B. 350

C. 480

D. 175

K&A #	295026 K1.01
Importance Rating	3.0 (RO) / 3.4 (SRO)

QUESTION 13

K&A Statement: **K1.01** - Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Pump NPSH

Justification:

A. A is incorrect – If the candidate fails to subtract 4.5 feet from the current Torus level or does not use the Fig. N1 calculation then the final overpressure will be 10.1 psig and the 10 psig curve would be used. At 200F in the Torus this would yield a single subsystem flow of approximately 240. This is a plausible distractor due to the answer being reasonably determined with the question stem and graph provided.

B. B is incorrect – If the candidate calculates the overpressure correctly but does not recognize that only one loop of Core Spray is available, at 200F in the Torus this would yield a two subsystem flow of approximately 350. This is a plausible distractor due to the answer being reasonably determined with the question stem and graph provided.

C. C is incorrect – If the candidate fails to subtract 4.5 feet from the current Torus level or does not use the Fig. N1 calculation then the final overpressure will be 10.1 psig and the 10 psig curve would be used. If the candidate also fails to recognize that only one loop of Core Spray is available with valve 40-12 closed, the two subsystem flow of approximately 480. This is a plausible distractor due to the answer being reasonably determined with the question stem and graph provided.

D. Correct D is correct – The calculated Torus overpressure uses the equation below Figure N1 on EOP-2. The final pressure is determined to be (9.5 feet – 4.5 feet = 5 feet x .433=2 psig + 6 psig=) 8psig, so 5 psig curve is used. The valve failures result in injection from Core Spray Subsystem #12 due to the closure of valve 40-12 which isolates all flow from the Core Spray Subsystem #11 pumps. With only one subsystem available the max flow is 175.

References: EOP-2 Fig. N1, C-19410-C Sht. 2 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2002 NRC Exam

Question History: NMP1 2002 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.8

QUESTION 14

Step DWT-2 in EOP-4, “Primary Containment Control” directs entry into EOP-2, “RPV Control” and verify the reactor is scrammed, BEFORE reaching 300F in the Drywell.

Which of the following lists why the reactor is scrammed before the bulk drywell temperature reaches 300F?

A. This reduces the rate at which heat is transferred from the RPV to the Drywell.

B. This will prevent exceeding the qualification of reactor vessel instrumentation cabling.

C. This will prevent RPV water level inaccuracies.

D. This ensures that Drywell Sprays will be effective.

K&A #	295028 A2.01
Importance Rating	4.0 (RO) / 4.1 (SRO)

QUESTION 14

K&A Statement: A2.01 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:
Drywell temperature

Justification:

A. CORRECT A is correct – Entry into EOP-2, "RPV Control" is performed in anticipation of containment spray initiation and a subsequent blowdown. This action reduces the rate of energy production and thus heat into the drywell (EOP Bases NER-1M-095, Rev. 2 page 174). Additionally, entry into EOP-2 allows the Operator to reduce reactor pressure which provides an additional means to reduce the heat transfer rate from the reactor to the Drywell.

B. B is incorrect – The performance of the reactor scram will not impact the environmental qualification of the cabling for reactor vessel instrumentation. The ADS system has components which are qualified for the 300F limit; however these are not reactor vessel instrumentation cables. This is a valid distractor because the basis for performance of the scram is tied to environmental qualification of equipment, however it is not the equipment listed in the answer.

C. C is incorrect – The RPV level inaccuracies that may be introduced during periods of high drywell temperature will not be corrected by the reactor scram. Additionally, the reduction in vessel pressure which normally accompanies the reactor shutdown could cause additional flashing or boiling of RPV reference legs and make the likelihood of developing level inaccuracies greater. This is a valid distractor for those candidates who recognize the relationship between drywell temperature and RPV level instrumentation but incorrectly assess the changing conditions as improving the likelihood for flashing to occur versus degrading with the high drywell temperature.

D. D is incorrect – As described in Answer B, the reactor scram has no direct effect on the effectiveness of the containment sprays. However this is a valid distractor for those candidates who assume that the reactor scram can be directly tied to initiation of containment sprays at a lower energy level in the containment and therefore improved effectiveness.

References: N1-SOP-40.2 Student Ref: None

Learning Objective: N/A

Question source: OC 2007 NRC Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.10

QUESTION 15

Unit 1 was operating at full power when it experienced a Main Turbine trip and an ATWS. The following conditions are present:

- Reactor Power is 15%.
- Reactor Level is being lowered in accordance with EOP-3 Failure to SCRAM.
- Reactor Pressure is 875 psig and is slowly rising
- Reactor pressure was initially stabilized between 875 and 1050 psig
- Reactor Pressure is to be maintained using ERVs due to issues with EC's
- Torus Temperature is 117°F and rising quickly.
- Torus Level is 11.24 feet and rising slowly

Determine the operational implications of these conditions on Reactor Pressure control.

A. an emergency blowdown is required for plant conditions

B. Reactor pressure must be maintained at or below 950 psig

C. Reactor pressure must be maintained at or above 950 psig

D. Continued use of the full Reactor pressure band is acceptable.

K&A #	295030 K1.03
Importance Rating	3.8 (RO) / 4.1 (SRO)

QUESTION 15

K&A Statement: K1.03 - Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat Capacity

Justification:

A. A is incorrect – Current plant conditions do not indicate an emergency blowdown is required for the current plant conditions.

B. CORRECT B is correct - Operation at the lower end of the Reactor Pressure band (below 950 psig) will result in maintaining Torus temperature below the HCTL Limit as Torus level continues to rise. Torus level is currently at the limit to transition to lower HCTL limits which will require operating a reduced reactor pressure to maintain the good (safe) side of the HCTL Limit. Depressurization during ATWS conditions when approaching HCTL is warranted and directed by EOP-4 and EOP-3 Step P-3.

C. C is incorrect – Operation at the upper end of the Reactor Pressure band (above 950 psig) will result in exceeding the HCTL Limit as Torus level continues to rise. Torus level is currently at the limit to transition to the lower HCTL limits and continued operation at the upper end of the pressure band will exceed HCTL above approximately 950 psig and direct the execution of a reactor blowdown.

D. D is incorrect – Continued use of the current pressure band will result in exceeding the HCTL curve as Torus level continues to rise. Torus level is currently at the limit to transition to the lower HCTL limits and continued operation at the upper end of the pressure band will exceed HCTL above approximately 950 psig and direct the execution of a reactor blowdown.

References:	N1-EOP-4, N1-EOP-3	Student Ref:	EOP-4 Fig. M

Learning Objective:	N/A	
Question source:	Modified PB 2005 Exam	
Question History:	None	
Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X
10CFR 55	41.10	

QUESTION 16

A plant transient has resulted in the following conditions:

- Reactor pressure 150 psig and steady
- No ERVs are open
- RPV water level -118" and steady
- Core Spray is injecting at 160 x 10^4 lbm/hr on Core Spray Loop 11 and 190 x 10^4 lbm/hr on Loop 12

Which one of the following is the condition of ADEQUATE CORE COOLING (ACC) with these conditions present?

A. There is no assurance of adequate core cooling

B. Steam Cooling without injection ensures adequate core cooling

C. Steam cooling with injection ensures adequate core cooling

D. Spray Cooling ensures adequate core cooling

K&A # 295031 A2.04
Importance Rating 4.6 (RO) / 4.8 (SRO)

QUESTION 16

K&A Statement: **A2.04** – Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Adequate Core Cooling

Justification:

A. CORRECT A is correct – Adequate Core Cooling cannot be assured because none of the mechanisms exist for ACC. RPV level is below Top of Active Fuel (TAF is -84 inches) thus eliminating submergence as a mechanism for adequate core cooling. Core spray flows from BOTH loops are not at or above the required 180 x 10^4 lbm/hr to establish adequate core cooling by spray cooling. Additionally RPV level is below Minimum Steam Cooling RPV Water Level (-109 inches) with injection from Core Spray. Spray Cooling without injection water level (-121 inches) is not usable under these conditions with Core Spray injection.

B. B is incorrect – Steam Cooling without injection is employed in EOP-9, Steam Cooling. With RPV water level below Minimum Steam Cooling RPV Water Level (-109 inches) and any injection source is lined up (one CRD pump) then ACC does not exist by Steam Cooling without injection, since EOP-9 Steam Cooling is not entered under these conditions. If entered override will direct exiting, with any injection source injecting. "ACC cannot be assured if RPV water level is below Minimum Steam Cooling RPV Water Level (-109 inches) and water is being injected into the RPV. (EOP Bases, EOP-9, page 281) This answer is a plausible distractor if the candidate believes the lower steam cooling level can be used with injection from Core Spray.

C. C is incorrect - Steam Cooling with injection is employed in EOP-2, RPV Control only if RPV water level is above Minimum Steam Cooling RPV Water Level (-109 inches), which it is not, with level at -118 inches. (EOP Bases, Definitions page 61). This is a plausible distractor for candidates that confuse the required steam cooling levels.

D. D is incorrect – Spray cooling is defined to exist when both Core Spray loops are injecting at or above 180 x 10^4 lbm/hr. The required flow is based on an analysis of a pipe whip event beyond the plant design basis. This is a plausible distractor due to one loop being above the limit and the other close to the required limit. This requirement is included in Step L-18 in EOP-2 as a limitation prior to executing an emergency blowdown.

References: EOP Bases NER-1M-095 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC Exam

Question History: NMP1 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 17

- All rods are not fully inserted following a reactor scram. N1-EOP-3 is entered. ARI is initiated. The following indications are on F Panel:

    - All of the blue SCRAM lights are extinguished
    - All of the amber accumulator lights are extinguished
    - All of the Scram Solenoid Group lights are extinguished
    - Reactor pressure is 375 psig
    - NO CRD pumps are available

Which one of the following methods in N1-EOP-3.1 could be used to complete the scram?

A. Venting the overpiston area

B. Pulling RPS fuses in 1S53 and 1S55

C. Repeated manual Scram Signals

D. Venting the scram air header

K&A #	295037 K3.07
Importance Rating	4.2 (RO) / 4.3 (SRO)

QUESTION 17

K&A Statement: K3.07 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Various alternate methods of control rod insertion

Justification:

A. A is incorrect – Per EOP-3.1 venting of the overpiston area with reactor pressure less than 400 psig may not cause the rod to insert and insertion via the reactor manual control system may be required. With reactor pressure at 375 psig and no CRD pumps available these options are no available.

B. B is incorrect – RPS has already de-energized as indicated by the Scram Solenoid Group lights being extinguished. The removal of fuses in 1S53 and 1S55 is performed to de-energize the RPS trip systems which have already been accomplished by the initial reactor scram.

C. C is incorrect – Repeated manual scram signals rely on the ability to reset and perform an additional scram to allow the forces impacted by the accumumlators and CRD to force the rods in. With the inability to complete either the initial or subsequent scram signals, there will be no additional forces applied to move the rods inward.

D. CORRECT D is correct – Venting of the scram air header will complete the scram action intended by both RPS and ARI initiation. The venting of the header will allow the HCU accumulators to commence and reactor pressure to complete the scram and insert the control rods. D is correct.

References: N1-EOP-3.1 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 18

EOP-6, Radioactive Release Control, contains a step to operate Turbine Building ventilation, as required.

Which one of the following is <u>NOT</u> one of the primary reasons for operating the Turbine Building ventilation system?

A. Prevent unmonitored ground level releases.

B. Capture leakage from the Reactor Building prior to release to the atmosphere.

C. Provide a controlled and elevated release point

D. Provide access to the Turbine building while the release is in progress

K&A # 295038 K1.02
Importance Rating 4.2

QUESTION 18

K&A Statement: K1.02 – Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Protection of the general public

Justification:

A. A is incorrect – Operation of the Turbine Building Ventilation system places the building under a negative pressure and limits the possibility of a ground level release. The exhaust is released at an elevated point via the plant stack. This minimizes the dose and impact to the public by minimizing the potential for a ground level release pathway. This is a plausible answer if the candidate does not recognize the bases for the step includes the desire to limit ground releases as part of the basis for restarting the ventilation system.

B. CORRECT B is correct – The Turbine Building ventilation system maintains a negative pressure in the Turbine Building to ensure releases from or through systems that pass through secondary containment is captured for release through the plant stack. However, leakage from the Reactor Building is not anticipated or designed to be captured by the Turbine Ventilation system operation. Although the Turbine Building does provide the area for blow-out panel releases to be received, it does not form a complete envelope around the reactor building to capture all leakage from the building.

C. C is incorrect. As described in EOP-6, the Turbine Building Ventilation system should be operated to direct any radioactivity released from the turbine building through an elevated, monitored path. This is a plausible distractor if the candidate does not recognize the primary flowpath of the turbine building exhaust is up the plant stack and past the radiation monitoring instrumentation for the stack.

D. D is incorrect – The restart of the Turbine Building HVAC system will help restore access to the Turbine Building as needed to support transient event mitigation. This is a plausible distractor if the candidate does not recognize the access requirements to the non-safety related structure as part of the bases for operation of the ventilation system.

References: EOP-6 Student Ref: None

Learning Objective: N/A

Question source: Modified VY 2007 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.10

QUESTION 19

Unit 1 is at 100% Power when the following occurs:

- A fire occurs in the Cable Spreading Room which initiates the fire suppression system
- The Fire Brigade reports the fire is out but there is heavy smoke filling the area

Which of the following actions are required prior to initiating smoke removal.

A. Manual reset of the Halon system is required prior to initiating purge

B. Notify Shift Manger to declare CREVS inoperable prior to initiating purge.

C. Manual reset of the CO2 system is required prior to initiating purge.

D. Manual reset of the smoke system for the cable spreading room is required prior to initiating purge.

K&A # 600000 A1.05
Importance Rating 3.1

QUESTION 19

K&A Statement: A1.05 – Ability to operate and/or monitor the following as they apply to PLANT FIRE ON SITE: Plant and control Room ventilation systems

Justification:

A. A is incorrect – The Cable Spreading Room is protected by CO2 not Halon. Therefore Halon is not required to be reset to remove smoke from the Cable Spreading Room. Plausible if the candidate does not remember which areas are protected by CO2 and which areas have Halon systems. Aux control room does have a halon system

B. B is incorrect – If the aux control room or control room is to be purged, CREVS must be declared inoperable because the smoke system will not isolate on high control room radiation signals. The Cable Spreading Room is in zone 4 and is not part of the aux control room or control room which are zone 5 and 6 and therefore this caution is not applicable. Plausible if the candidate thinks that the cable spreading room is part of the Aux control room and control room zones.

C. CORRECT – C is correct. Per system description if the CO2 system initiates then fire dampers close and smoke fans stop. This must be manually reset on Local fire Panel 1 to be able to start the smoke system.

D. D is incorrect – The smoke system reset switch is used to shutdown the system per section G4.0. Plausible if the candidate thinks that the system will need to be reset prior to using.

References: N1-OP-21F Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 2002 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 20

The plant is operating at 100% during degraded grid conditions. Generator conditions are as follows:

- 640 MWe
- 200 MVAR
- H2 Pressure is 45 psig

At 1450 the following alarms occur:

- Alarm A1-4-1 GENERATOR H2 SEAL OIL PRESSURE LOW annunciates
- Alarm A1-4-2 GENERATOR HYDROGEN SYSTEM annunciates
- Computer point D113 GEN H2 SYS SUP OIL PR _ LOW has been received
- Generator H2 pressure is verified at 28 psig and dropping slowly
- The MSOP is verified to still be in service

What is the first action required to be taken under these conditions?

A. SCRAM the reactor per N1-SOP-1 and Verify Turbine Trip per N1-SOP-31.1

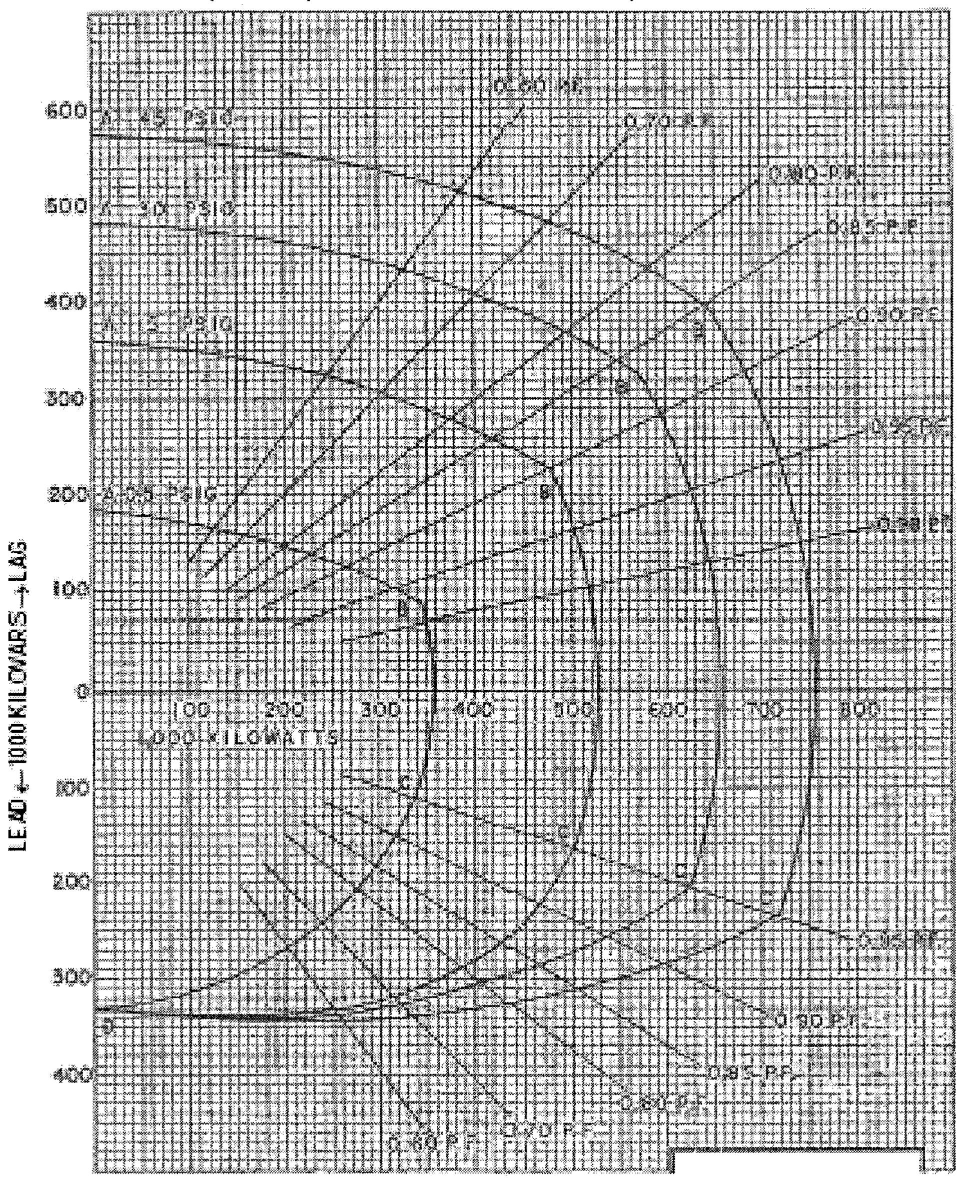
B. Commence an Emergency Power Reduction per N1-SOP-1.1

C. Bypass the Seal Oil Pressure Regulator per N1-SOP-32, Attachment 2

D. Add H2 to the Generator per N1-OP-7

ATTACHMENT 1: ESTIMATED CAPABILITY CURVE

ATB 4 POSE, 775,000 KVA, 1800 RPM, 24,000 VOLTS
0.85 P.F., 0.58 SCR, 45 PSIG HYDROGEN PRESSURE, 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

K&A # 700000 G2.1.25
Importance Rating 3.9 (RO) / 4.2 (SRO)

QUESTION 20

K&A Statement: **G2.1.25** – Ability to interpret reference materials such as graphs, curves, tables, etc. as it relates to **GENERATOR VOLTAGE AND ELECTRICAL GRID DISTURBANCES.**

Justification:

A. A is incorrect – Under the conditions described, entry into N1-SOP-32, Generator Auxiliaries Failures, is required. This is a plausible distractor because if H2 gas is dropping rapidly with the given plant conditions, a reactor scram and turbine trip would be required. Given the conditions that H2 pressure is dropping slowly a reactor scram is not warranted and this answer is incorrect.

B. CORRECT B is correct – Under the conditions described, entry into N1-SOP-32, Generator Auxiliaries Failures, is required. In executing the immediate actions in SOP-32, the immediate action requires evaluation of the current conditions against the capability curve in Attachment 1 of SOP-32. Current plant conditions place the generator outside the Estimated Capability curve for H2 pressure of 28 psig and require an emergency power reduction per N1-SOP-1.1 until operating within the curve.

C. C is incorrect – Bypassing of the Seal Oil pressure regulator is a required action of N1-SOP-32 and ARP HSC 1 to help maintain pressure. This is a plausible distractor for those candidates that do not recognize the entry conditions to SOP-32 and use ARP guidance to answer the question or those that incorrectly read the generator conditions as being on the safe side of the operating curve.

D. D is incorrect – Adding H2 to the generator is a follow up action of SOP-32 and is also directed from ARP A1 4-2 to restore pressure. This is a plausible distractor for those candidates that do not recognize the entry conditions to SOP-32 and use ARP guidance to answer the question or those that incorrectly read the generator conditions as being on the safe side of the operating curve.

References: N1-SOP-32 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.10

QUESTION 21

Given these initial conditions:

- NMP Unit 1 is operating at 100% power with six drywell cooling fans running
- RBCLC pump 12 and 11 are running

Then the following events occur:

- The feeder breaker to PB161A from PB16A trips on overcurrent
- Drywell (DW) pressure begins to rise
- No operator action is taken

Which one of the following explains the reason for the Drywell pressure rise, based on the above conditions?

A. Three DW air coolers tripping and DW temperature increase

B. The loss of a RBCLC pump and DW temperature increase

C. The RBCLC to the DW Containment Isolation Valve shutting

D. The loss of a RBCLC pump and three DW air coolers tripping

K&A # 295010 K2.05
Importance Rating 3.7

QUESTION 21

K&A Statement: **Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Drywell Cooling and ventilation.**

Justification:

A. Correct –The Power to 3 of the DW coolers (14,15 and 16) is PB 161A.

B. Incorrect but plausible if the applicant does not know that the RBCLC pumps are not powered by PB 161A.

C. Incorrect but plausible if the applicant does not know that the DW Containment isolation valve is DC powered.

D. Incorrect but plausible if the applicant does not know that the RBCLC pumps are not powered by PB 161A.

References: N1-OP-8, N1-OP-30 Student Ref: None

Learning Objective: N/A

Question source: NMP 1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 22

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

- Annunciator L1-4-4 Drywell-Torus Temp is in Alarm
- Drywell Temp Avg CH 11 is 140 degrees and trending higher
- Drywell Temp CH 12 Elev 319 is 180 degrees and alarming
- Drywell Temp CH 11 Elev 230 is 140 degrees and alarming
- Drywell Pressure is 1.2 psig and steady
- Drywell Unidentified Leakage is .22gpm and steady
- Six Drywell Fans are in service
- Service Water is valved into 11 RBCLC Heat Exchanger

What operator actions are required?

A. Enter N1-EOP-4 Primary Containment Control.

B. Obtain Shift Manager permission to bypass alarming points. Increase frequency of monitoring of Drywell Temperatures.

C. Take manual Control of 70-23B and lower RBCLC Temperature.

D. Reduce Reactor Power per N1-SOP-1.1 Emergency Power Reduction

K&A # 295012 A1.02
Importance Rating 3.8

QUESTION 22

K&A Statement: **Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell Cooling System.**

Justification:

A. Incorrect but plausible, entry condition is > 150 degrees on average temperature. Only elevation 319 is >150.

B. Incorrect but plausible, only if a single elevation is alarming can the alarm be bypassed per N1-OP-8.

C. Correct – per step 3.2 of N1-OP-8 lower RBCLC by taking manual control of 70-23B

D. Incorrect but plausible, Lowering Rx Power will reduce the heat load on the Drywell, however with these conditions a rapid power reduction is not called for.

References: N1-OP-8, N1-OP-11, N1-ARP L1-4-4 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 23

From normal full power operation, which of the following will result in a stable higher power level?

A. Inadvertently isolating the Reactor Water Cleanup System.

B. Inadvertently opening an ERV.

C. Main Condenser Circulating Pump Trip.

D. An Extraction Steam Non-return valve trips on high feedwater heater level.

K&A # 295014 AK2.07
Importance Rating 3.9

QUESTION 23

K&A Statement: Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: Reactor Power.

Justification:

A. Incorrect because the reactor power level will be lower due to not having to heat the cleanup flow return to the reactor.

B. Incorrect because the lower reactor pressure will cause reactor power to lower, the EPR will respond to lower steam flow to the turbine and the feedwater system will lower feedwater flow to stabilize reactor level.

C. Incorrect because the circulating water pump trip will cause an increase in condensate temperature and therefore a higher feedwater temperature and a lower power level.

D. Correct – The steam flow to the heater is isolated causing colder feedwater to enter the reactor causing reactor power to increase.

References: Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 24

Given the following conditions:

- An ATWS is in progress.
- Reactor power is 18%
- Reactor Pressure is 930 psig and is being controlled with the main turbine bypass valves.
- RPV level has been intentionally lowered to -41" in accordance with step L-9 EOP-3, “Failure To Scram”

WHICH ONE of the following explains the reason for this RPV level reduction?

A. Concentrate boron in the core

B. Increase void fraction inside the shroud

C. Increase preheating of the incoming feedwater

D. Reduce natural circulation driving head through the core

K&A # 295015 Incomplete SCRAM

Importance Rating 3.8

QUESTION 24

K&A Statement: G2.4.9-Knowledge of low power/shutdown implications in accident mitigation strategies, as it relates to Incomplete SCRAM

Justification:

A. Incorrect. Plausible because reducing feed will limit dilution.

B. Incorrect. Plausible because void fraction could increase due to lowered head of the column of water.

C. Correct –Per EOP 3 Failure to scram basis, Element L-5 override states the feedwater sparge is at -17 inches and uncovering the sparger heats the incoming feedwater, thereby reducing the subcooling at the core inlet. A level 2 ft below the feedwater sparger is low enough to reduce subcooling by 65-75%. .

D. Incorrec. Plausible because driving head will decrease with a lower water level.

References: EOP-3 Basis Student Ref: EOP-3

Learning Objective: N/A

Question source: LIM 08 NRC Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55.41

QUESTION 25

Unit 1 is starting up following refueling outage with preparation for the Initial Motor Driven Feedwater Pump Start per N1-OP-16 in progress.

The following events then occurred at the noted times:

09:30 F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB
F3-1-5, CRD CHARGING WTR PRESSURE HI/LO

09:34 F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, is received for Control Rod 10-19 accumulator pressure.

09:36 CRD Pump 12 did not start when its control switch was placed to START

09:37 F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, received for Control Rod 34-35 accumulator pressure.

Which one of the following is LATEST TIME to insert a manual scram while attempting to restore CRD?

A. 09:36

B. 09:56

C. 09:34

D. 09:54

K&A # 295022 A2.01
Importance Rating 3.5 (RO) / 3.6 (SRO)

QUESTION 25

K&A Statement: **A2.01** - Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: **Accumulator pressure**.

Justification:

A. A is incorrect – This is a plausible distractor for those candidates that believe that both pumps need to be verified to be non-operational prior to scramming the plant.

B. B is incorrect – This answer is provided for those candidates that incorrectly assess reactor pressure to be above 900 psig given the initial conditions. This is a plausible distractor for those candidates that believe that both pumps need to be verified to be non-operational prior to starting the 20 minute clock for restarting at least one CRD pump. The start of the 20 minute completion time is independent of whether or not a start of the other pump has been attempted.

C. CORRECT C is correct – The initial condition of preparing to place the first motor driven feedpump in service requires the reactor between 300 and 350 psig per Step 3.16 of N1-OP-43A, Plant Startup. With reactor pressure at approximately 300 psig, an immediate scram is required after the trip of the CRD pump and receipt of the 1st accumulator trouble alarm.

D. D is incorrect – With reactor pressure above 900 psig, 20 minutes are permitted from receipt of the first accumulator trouble alarm with no CRD pumps running until the scram is required. With reactor pressure below 900 psig, a scram must be inserted immediately upon receipt of the first accumulator trouble alarm if no CRD pumps are running.

References: N1-SOP-5.1 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2005 NRC Exam

Question History: NMP1 2005 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.10

QUESTION 26

During preparation for the upcoming refueling outage new fuel is being moved in the spent fuel pool per N1-FHP-24, Movement of Fuel and Blade Guides Within The Spent Fuel Pool.

During the moves, the grapple failed and the control room is notified that the new fuel fell on irradiated fuel bundles and caused damage to the irradiated bundles.

The following alarms are received in the Control Room:

- Area Radiation Monitors (H1 4-8)
- F306 RB340 REFUEL LO RNG is displaying 900 MR/H
- F307 RB340 REFUEL HI RNG is displaying 900 MR/H
- F316 RB340 East Wall is displaying 25 MR/H
- React Bldg Vent Rad Monitor Off Normal
- E476 RB VENT RMON 11 is displaying 3.8 MR/H
- E477 RB VENT RMON 12 is displaying 5.0 MR/H
- F307 RB340 REFUEL HI RNG is displaying 900 MR/H

Evaluate these conditions and determine the expected ventilation response.

A. Reactor Bldg. Ventilation Trips due to RB Vent Rad Monitor #12
RBEV initiates and aligns to the Reactor Building

B. Reactor Bldg. Ventilation Trips due to Refuel Bridge High Range Rad Monitor
RBEV initiates and aligns to the Reactor Building

C. Reactor Bldg. Ventilation continues to run due to trip setpoint not being reached
RBEV remains in standby

D. Reactor Bldg. Ventilation continues to run due to Refuel Bridge High Range Monitor Bypass Switch is maintained in BYPASS for new fuel movement in the Spent Fuel Pool.
RBEV remains in standby

K&A #	295034 K3.02
Importance Rating	4.1 (RO) / 4.1 (SRO)

QUESTION 25

K&A Statement: K3.02 – Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Starting SBGT

Justification:

A. CORRECT A is correct – RBEV will initiate on a single channel initiation from either RPS channel #11 or #12 or from the Refuel Bridge High Range rad monitor through the channel 12 trip logic. The trip setpoint for Reactor Building Ventilation isolation and RBEV initiation is 5.0 mR/h on the Reactor Building ventilation process radiation monitors and 1000 mR/h from the Refuel Bridge High Range monitor. The trip setpoint has been reached on RB Vent Rad Monitor 12 and will result in a trip of the Reactor Building ventilation system and start of the RBEV system.

B. B is incorrect – RBEV will initiate on a single channel initiation from the Refuel Bridge High Range Rad monitor through the channel 12 trip logic. The trip setpoint for Reactor Building Ventilation isolation and RBEV initiation is 1000 mR/h from the Refuel Bridge High Range monitor. The trip setpoint has NOT been reached on Refuel Bridge High Range Rad Monitor. Therefore the answer is incorrect. Plausible because the Bridge High Range monitor does start RBEV, however in this case the setpoint has not been reached.

C. C is incorrect. - The trip setpoint for Reactor Building Ventilation isolation and RBEV initiation is 5.0 mR/h on the Reactor Building ventilation process radiation monitors. The trip setpoint has been reached on RB Vent Rad Monitor 12 and will result in a trip of the Reactor Building ventilation system and start of the RBEV system. Therefore the answer is incorrect. Plausible if the candidate does not know the setpoint has been reached.

D. D is incorrect – RBEV will initiate on a single channel initiation from the Refuel Bridge High Range Rad monitor through the channel 12 trip logic. The trip from the Refuel Bridge High Range Rad Monitor can be bypassed with the use of the Refuel Bridge High Range Monitor Bypass Switch. This switch is required to be maintained in the REFUEL position when placing new fuel in the spent fuel pool per N1-FHP-9 or moving fuel in the spent fuel pool per N1-FHP-24. This enables the trip of the Refuel Bridge High Range Rad Monitor for the evolution described in the stem. Therefore this answer is incorrect. Plausible if the candidate thinks that when moving new fuel to the fuel pool that the trip is bypassed.

References:	EOP-4, N1-FHP-9, N1-FHP-24, H1 4-8, L1 4-3	Student Ref:	None

Learning Objective:	N/A	
Question source:	Modified PB 2005 Exam	
Question History:	None	
Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X
10CFR 55	41.7	

QUESTION 27

An unisolable RWCU leak has resulted in the SE Containment Spray Room with a water level above the Maximum Safe Value. The current level is 64 inches and continuing to rise. EOP-5 requires a scram to be inserted.

Which one of the following identifies the correct EOP Bases for inserting the scram?

A. Ensure reactor is shutdown prior to leak getting larger.

B. Places the primary system in its lowest possible energy state.

C. Allow personnel into secondary containment to perform safe shutdown actions

D. Reduce the break flow into secondary containment.

K&A #	295036 G2.4.18
Importance Rating	3.3 (RO) / 4.0 (SRO)

QUESTION 27

K&A Statement: G2.4.18 – Knowledge of the specific bases for EOPs: Secondary Containment High Sump / Area Water Level

Justification:

A. A is incorrect – Although the energy provided by the primary system discharging into secondary containment is directly effected by pressure and indirectly by power level there is no direct correlation nor guarantee that a primary system leak will get larger with time as this answer suggests. This answer is plausible due to the routine actions taken in the simulator to shut down the reactor with leaks of various sizes and locations. The understanding that this is done to limit the impact of the existing leak versus the potential growth of the leak makes this answer plausible.

B. B is incorrect – Placing the primary system in its lowest possible energy state is accomplished by and the bases for a blowdown of the reactor. The reactor in a scrammed condition is in a lower energy state than an online reactor but is not in the lowest energy state achievable due to the primary system remaining at pressure. This answer is plausible if the candidate does not recognize the reactor is not in it's lowest energy state without depressurizing the reactor vessel. Since there is no discussion of reactor pressure or potential for vessel blowdown, this answer is plausible.

C. C is incorrect. - The answer provided is part of the definition for establishing the maximum safe value. The "Maximum Safe Value is defined to be the highest value at which:

- Equipment necessary for the safe shutdown of the plant will operate, and
- Personnel can perform any actions necessary for the safe shutdown of the plant.

This answer is plausible since the Maximum Safe Value is used a supporting value and basis for the scram. The interrelationship of the Maximum Safe Value definition and the basis for the reactor scram makes this answer a plausible distractor.

D. CORRECT D is correct – The bases for the scram is defined in the EOP bases as follows: If a parameter exceeds its maximum safe operating value, plant safety may be jeopardized. If a primary system is known to be discharging into the secondary containment but it cannot be isolated, the reactor is scrammed and EOP-2 entered (if not already in use) "before" any parameter reaches a Maximum Safe Value.

- ***A scram reduces*** the rate of energy production and thus the heat input, radioactivity release, and ***break flow into the secondary containment***
- If possible, the reactor should be shut down before a blowdown is performed

D is the correct answer.

References: NER-1M-095 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.10

QUESTION 28

The plant is shutdown for a refueling outage. Shutdown Cooling has recently been placed in service. The following are plant conditions.

- 2 SDC pumps are in operation
- 2 RBCLC heat exchangers are in service
- 5 Recirculation pumps are in operation
- Recirculation loop temperatures are 330 F and slowly lowering
- Reactor Level is +70 inches and slowly lowering
- SDC flow is 4000 gpm

Later in the shift:

- The 5 Recirculation loops temperatures are 345 F and slowly rising
- Reactor level is +60 inches and slowly lowering

Which one of the following explains the rise in Recirculation loop temperatures?

A. Shutdown Cooling System has isolated on high temperature.

B. The SDC pumps do not have the minimum Net Positive Suction Head.

C. Total Shutdown Cooling flow is too low to ensure adequate core circulation.

D. Two SDC heat exchangers are inadequate to remove the decay heat generated shortly after shutdown.

K&A # 205000 A1.03
Importance Rating 3.3

QUESTION 28

K&A Statement: Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM controls including: Recirculation loop temperatures.

Justification:

A. Incorrect because the SDC system isolates at 350 F. Plausible because an isolation of the system would stop decay heat removal.

B. Correct – With 2 pumps running the SDC pumps need >65 inches for NPSH.

C. Incorrect because the circulation through the core is adequate if recirculation pumps are running. Plausible because without recirculation pumps and SDC flow too low, thermal stratification could occur.

D. Incorrect but plausible if the applicant does not know that the SDC system was designed to go to cold shutdown with 2 pumps and heat exchangers.

References: N1-OP-4 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5

QUESTION 29

- The plant is operating at 100% power with #11 and #13 Feedwater pumps maintaining reactor level and #12 Feedwater pump is available.

Following a reactor scram and turbine trip the following conditions exist:

- RPV pressure 900 psig.
- RPV water level 65 inches and steady.

Which one of the following failures would result in the current plant conditions?

A. #12 Feedwater Pump Discharge Pressure instrument has failed high.

B. Feedwater Flow instrument has failed high.

C. #12 Feedwater Pump Discharge Pressure instrument has failed low.

D. #11 Feedwater Pump Discharge Pressure instrument has failed low.

K&A # 206000 K6.11
Importance Rating 3.6

QUESTION 29

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM: Nuclear Boiler Instrumentation.

Justification:

A. Incorrect. Plausible if applicant does not know #12 discharge pressure failing high will not prevent the #12 FCV from opening.

B. Incorrect. Plausible because the feedwater flow failing high would cause the #11 FCV to not open, however the level is controlling at the #11 setpoint..

C. Correct –the reactor level is being controlled at the #11 setpoint of 65 inches. This means the #12 pump is not injecting and not controlling at its setpoint of 72 inches. The #12 FCV is shut due to when the pump is idle, the controller output is held at a minimum until 990 psig is produced at the feedpump.

D. Incorrect but plausible, #11 FCV would be closed if the discharge pressure failed low, however the level is controlling at the #11 setpoint.

References: N1-OP-16 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 30

NMP 1 was manually scrammed due to a steam leak in the primary containment. Current plant conditions are as follows:

- Drywell Pressure is 5 psig and rising.
- FW pump 11 and 12 were operating in the HPCI mode and injected until they both tripped on high level.
- Reactor Water Level is +96 inches and dropping slowly.
- Reactor Pressure is 920 psig and being controlled automatically by the EPR.

The Feedwater System:

A. Will start the 11 and 12 FW pumps and the pumps will inject automatically when Reactor Level reaches +53 inches.

B. Will start the 11 and 12 FW pumps and the pumps will inject automatically when Reactor Level reaches +90 inches.

C. Will not restart automatically. The operator must manually reset the high level trip.

D. Will start the 12 FW pump and the pump will inject automatically when Reactor Level reaches +53 inches.

K&A # 206000 K4.04
Importance Rating 4.0

QUESTION 30

K&A Statement: Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Resetting system isolations.

Justification:

A. Correct –Both running pumps will get an auto start when level lowers to +53 inches, low level scram setpoint.

B. Incorrect but plausible if the applicant thinks that the pumps will start once the high level trip resets.

C. Incorrect but plausible if the applicant thinks that the high level trip needs to be reset manually to have the pumps auto start on the low level scram trip.

D. Incorrect but plausible if the applicant thinks that only the preferred pump #12 will start.

References: N1-OP-16 Student Ref: None

Learning Objective: N/A

Question source: PB 2007

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 31

The reactor was operating at 100% power when the following occurred:

- MSIV closure and a reactor scram
- Emergency Cooling System initiated
- Emergency Cooling Loop 11 auto isolates
- Emergency Cooling Channel 11 bypass switch is placed in the BYPASS position
- Reactor pressure has been >1100 psig for 20 seconds

With no operator actions, which one of the following describes the expected positions of the Emergency Cooling system valves?

	EC Condensate Return Valve 11	EC Steam IV 111
A.	Closed	Closed
B.	Open	Closed
C.	Closed	Open
D.	Open	Open

K&A # 207000 K1.11
Importance Rating 3.4

QUESTION 31

K&A Statement: Knowledge of the physical connections and/or cause effect relationships between ISOLATION (EMERGENCY) CONDENSER and the following: Primary containment Isolation system.

Justification:

A. Incorrect because the condensate return valve will auto open with the bypass signal bypassed. Plausible if the operator thinks that the Condensate return valve needs to be manually opened.

B. Correct – The condensate return valve will open because there is an open signal and the isolation signal has been bypassed. The Steam IV can be manually opened but will not auto open.

C. Incorrect because the condensate return valve would be open and the steam valve closed. Plausible if the operator thinks the bypass switch will return both valves to a normal lineup.

D. Incorrect because the steam valve will not auto open. Plausible if the operator thinks that bypassing the isolation signal will return the steam valve to its normal position of open.

References: Student Ref: None

Learning Objective: N/A

Question source: NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.2 to 41.9

QUESTION 32

The plant has had a small break LOCA.

Plant Conditions are as follows:

- Plant has scrammed
- Drywell Pressure is 4.5 psig
- Reactor Water Level is -20 inches
- Channel 11 ADS reset Pushbutton is NOT illuminated
- Channel 12 ADS reset Pushbutton is illuminated

What is the status of the Core Spray System two minutes later assuming no operator actions taken?

A. All Core Spray pumps are running and after 115.5 seconds the backup ERV's opened allowing Core Spray to Inject into the vessel.

B. All Core Spray pumps are running with minimum flow to the Torus.

C. All Core Spray pumps are running and after 111 seconds the primary ERV's opened allowing Core Spray to Inject into the vessel.

D. All Core Spray pumps are running and after 115.5 seconds the primary and backup ERV's opened allowing Core Spray to Inject into the vessel.

K&A # 209001 K6.11
Importance Rating 3.6

QUESTION 32

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the Low Pressure Core Spray System: ADS.

Justification:

A. Incorrect because the backup ERV's will not open if the primary ERV's open. Plausible if the applicant thinks the channel 12 timer controls only the backup ERV's, however each timer controls both primary and backup ERV's.

B. Incorrect because although the channel 11 timer is not energized, ADS will actuate with only one channel timed out. Plausible if applicant thinks that need both timers to open ERV's.

C. Correct – Only one timer needs to time out to actuate the ADS valves. Channel 12 ADS reset pushbutton illuminated indicates that the channel 12 timer is energized.

D. Incorrect because if the primary valves open the backup valves will not open. Plausible if applicant does not know that backup valves only open if primary valves fail to open.

References: ADS Lesson Plan Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 33

A plant event has resulted in a reactor scram and a loss of Reactor Protection Bus 11. Degrading containment conditions has resulted in the following:

- Reactor Pressure is at 305 psig
- Drywell pressure is at 3.7 psig.

Based on the above, how will Core Spray be affected by the Reactor Protection Bus 11 loss?

A. All Core Spray pumps will start and All injection valves will open.

B. 111 and 112 Core Spray and Core Spray Topping Pumps, and Inside Isolation Valves 40-10 and 40-11 will open.

C. No Core Spray pumps will start and No Injection valves will open.

D. 121 and 122 Core Spray and Core Spray Topping Pumps, and Inside Isolation Valves 40-09 and 40-01 will open.

K&A # 209001 K2.03
Importance Rating 2.9

QUESTION 33

K&A Statement: Knowledge of the electrical power supplies for the LOW PRESSURE CORE SPRAY SYSTEM to the following: Initiation Logic.

Justification:

A. Correct – Core Spray Logic is a de-energize to function logic. Therefore, the loss of an RPS bus produces a half initiation signal. The other system will initiate on a valid signal, which the drywell pressure is greater than 3.5 psig and the valves will open <365 psig in the reactor.

B. Incorrect - because the failure will not prevent an initiation signal. Plausible if the candidate thinks that the "B" Core spray loop lost initiation logic.

C. Incorrect - because the failure will not prevent an initiation signal. Plausible if the candidate thinks that the loss of RPS bus 11 will prevent getting a valid core spray signal.

D. Incorrect - because the failure will not prevent an initiation signal. Plausible if the candidate thinks that the "A" Core spray loop lost initiation logic.

References: C19859 SHt 9 Student Ref: None

Learning Objective: N/A

Question source: VY07

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 34

The plant was operating at 100% power, current plant conditions are as follows:

- A containment Isolation has occurred
- Reactor Power is 15%
- ERV's are cycling AUTOMATICALLY to control reactor pressure

The #11 Liquid Poison Pump has been manually started from the Control Room.

Indication that the squib has fired and the explosive valve is open are:

A. Current meter on Panel 1S-65 in Auxiliary Control Room Indicates .15 amps.

B. Discharge Pressure of Liquid Poison Pump is 1100 psig.

C. Current meter on Panel 1S-65 in Auxiliary Control Room indicates 2 amps.

D. Discharge pressure of Liquid Poison Pump is 1500 psig.

K&A # 211000 K5.04
Importance Rating 3.1

QUESTION 34

K&A Statement: Knowledge of the operational implications of the following as they apply to Standby Liquid Control System: Explosive valve operation.

Justification:

A. Incorrect but plausible, meter indicates there is continuity and valve has not fired.

B. Correct – Liquid poison is injecting because the discharge of the pump is slightly above the ERV pressure setpoint.

C. Incorrect but plausible, 2 amps is normal firing current, but after firing indication goes to 0 amps due to loss of continuity.

D. Incorrect but plausible, valve is not open, pump is at dead head pressure of pump.

References: N1-ARP-K1-2-1 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5

QUESTION 35

A plant startup is in progress.

- MODE switch is in STARTUP.
- All IRM's are on Range 8.
- All MSIV's are open.
- Reactor Pressure is 200 psig.

When the operator moves the IRM Range switches to the next range, he moves one RPS Channel 11 IRM to Range 10 and one RPS Channel 12 IRM to Range 10.

What is the plant response and the reason for the plant response?

A. All MSIV's remain open due to the <850 PSIG closure of the MSIV's is Bypassed in the STARTUP Mode.

B. All MSIV's close, however the reactor remains operating due to the 600 PSIG bypass of the MSIV closed Trip signal.

C. All MSIV's remain open due to the <850 PSIG closure of the MSIV's is Bypassed by the combination of the MODE switch in Startup and the current IRM Range switch positions.

D. All MSIV's close and the Reactor Scrams due to the MSIV closed Trip signal.

K&A # 212000 K5.02
Importance Rating 3.3

QUESTION 35

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to RPS: Specific Logic Arrangements

Justification:

A. Incorrect because with the mode switch in startup the IRM's need to be on range 1-9 for the MSIV closure to be bypassed. Plausible if the applicant thinks that the bypass is only mode switch dependant.

B. Incorrect because the MSIV's will close and the trip is not bypassed by the <600 psig bypass because of the IRM's being on range 10. Plausible if the candidate thinks that the 600 psig bypass is in effect at all times when < 600 psig.

C. Incorrect because with the mode switch in startup the IRM's need to be on range 1-9 for the MSIV closure to be bypassed. Plausible if the applicant thinks that it takes more than 1 IRM in an RPS channel to be in Range 10 to interrupt the bypass feature.

D. Correct – The bypass of the <850 PSIG MSIV closure is interrupted in each RPS channel by taking an IRM to range 10. Therefore the MSIV's will close. The <600 PSIG bypass of the MSIV scram is also interrupted by the IRM's in range 10.

References: RPS Drawing C19859 sht 2,3,5,6 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5

QUESTION 36

The reactor was operating at 100% power.

The Reactor Protection System (RPS) has just received a valid trip signal on both channels '11' and '12' (Reactor Water Level Low).

Which ONE (I) of the following correctly describes how RPS functions to insert control rods?

A. Both backup scram valve solenoids de-energize, which then vents instrument air pressure via the scram pilot valves directly to atmosphere, causing all control rods to be rapidly inserted (scrammed) by HCU accumulator and reactor pressure.

B. Both sets of scram pilot valve solenoids are energized allowing instrument air to be vented from the scram pilot air header. This permits the scram valves to open, causing all control rods to be rapidly inserted (scrammed) by HCU accumulator and reactor pressure.

C. Both backup scram valve solenoids energize, which then vents instrument air pressure via the backup scram valves from the scram pilot valve air header, causing all control rods to be rapidly inserted (scrammed) by HCU accumulator and reactor pressure.

D. Both sets of scram pilot valve solenoids are de-energized, allowing instrument air to be vented from the scram valves. This permits the scram valves to open, causing all control rods to be rapidly inserted (scrammed) by HCU accumulator and reactor pressure.

K&A # 212000 K4.08
Importance Rating 4.2

QUESTION 36

K&A Statement: Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and /or interlocks which provide for the following: Complete Control Rod insertion Following Scram Signal Generation.

Justification:

A. Incorrect but plausible, the air is vented from the scram pilot air header directly through the backup scram valves for the backup scram function, vice directly to atmosphere.

B. Incorrect but plausible because scram pilot valve solenoids are de-energized to actuate, not energized.

C. Incorrect but plausible because both backup scram valve solenoids have to de-energize to scram rods, not energized.

D. Correct - Scram pilot valve solenoids are de-energized, allowing instrument air to be vented from the scram valves. This permits the scram valves to open, causing all control rods to be rapidly inserted (scrammed) by HCU accumulator pressure and reactor pressure.

References: Student Ref: None

Learning Objective: N/A

Question source: Pilgrim 2007 NRC exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 37

The plant is starting-up after a 5-day forced outage. The following conditions exist:

The MODE Switch is in STARTUP, with control rod withdrawals in-progress

- IRMs 11, 12, 15,16, 18 read 72-74 on Range 2
- IRMs 13,14, and 17 read 9 - 10 on Range 3

A malfunction in the IRM drive circuitry caused IRM 13 to withdraw to the full-out position.

Which of the following states the effect on the plant and the required Operator actions to continue withdrawing control rods?

A. This will result in panel annunciators **ONLY**; withdrawing control rods may continue without any other control panel manipulations.

B. This will result in panel annunciators and a rodblock from IRM downscale **ONLY**; bypassing the IRM is required to continue withdrawing control rods.

C. This will result in panel annunciators and a rodblock from IRM downscale AND IRM detector position; bypassing the IRM is required to continue withdrawing control rods.

D. This will result in panel annunciators, a rodblock and a 1/2 scram; bypassing the IRM and resetting the 1/2 scram is required to continue withdrawing control rods

K&A # 215003 K3.02
Importance Rating 3.6

QUESTION 37

K&A Statement: Knowledge of the effect that a loss or malfunction of the Intermediate Range Monitor (IRM) System will have on the following:Reactor manual control.

Justification:

A. Incorrect because it does not list rodblocks which need to be bypassed.

B. Incorrect because it does not list all rodblocks.

C. Correct – The following provide IRM rod blocks: IRM downscale, detector not inserted. When the IRM comes off the full inserted it will cause a rod block. The IRM will also go downscale as the detector is withdrawn.

D. Incorrect because the downscale and detector not inserted are rodblocks only and not scram inputs. Plausible if operator thinks IRM downscale gives ½ scram signal and rod block signal.

References: N1-OP-38B   Student Ref: None

Learning Objective: N/A

Question source: OC 2006

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 38

A loss of 24VDC occurs to Nuclear Inst Bus 112.

Which of the following describes the effect on Nuclear Instruments?

	SRM's	IRM's	APRM's
A.	No change	Fails Low	No change
B.	Fails Low	Fails Low	No change
C.	Fails Low	Fails Low	Fails Low
D.	Fails Low	No change	No change

K&A # 215003 K2.01
Importance Rating 2.5

QUESTION 38

K&A Statement: Knowledge of Electrical Power Supplies to the following: IRM channels/detectors.

Justification:

A. Incorrect because SRM's will fail low.

B. Correct – Loss of 24VDC causes SRM's and IRM's to fail downscale. APRM's are not powered from 24VDC.

C. Incorrect because APRM's are not powered from 24VDC. Plausible because in the plant radiation monitors and IRMS's and SRM's are powered from 24 VDC. Apllicant may think that APRM's are also powered from 24 VDC.

D. Incorrect because IRMS will fail low.

References:	N1-OP-47B, C-22024-C, sht 3 and 4	Student Ref:	None
Learning Objective:	N/A		
Question source:	HC 2005 NRC		
Question History:	None		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	 X	
10CFR	41.7		

QUESTION 39

- A reactor startup is in progress. The reactor is critical and a heatup is in progress. SRM's are partially withdrawn. The following conditions exist:

- SRM 11 reads 175 cps
- SRM 12 reads 140 cps
- SRM 13 reads 120 cps
- SRM 14 reads 220 cps
- Refuel Instrument Trip Bypass 11 is in Non-Coincident
- Refuel Instrument Trip Bypass 12 is in Non-Coincident
- Electrical power is lost to SRM 13 High Voltage Power supply

What is the effect of the loss of SRM 13 High Voltage Power Supply and what would be the necessary actions to continue the startup?

A. Rod Block due to loss of voltage. Bypass SRM 13 and continue with the startup.

B. SRM 13 fails downscale. Declare SRM 13 inoperable and continue the startup.

C. SRM 13 fails upscale and causes a half scram. Bypass SRM 13, reset the half scram and continue with the startup.

D. Rod Block due to loss of voltage. Suspend startup until repairs are complete.

K&A # 215004 G.1.27
Importance Rating 3.9

QUESTION 39

K&A Statement: Knowledge of the purpose and function of major system components and controls as related to Source Range Monitors.

Justification:

A. Correct – With a low detector voltage and the mode switch in startup, the SRM 13 will give a rod block. The other 3 SRMs are operable and per tech specs, startup can continue with 3 operable. Need to bypass the SRM to clear the rod block.

B. Incorrect because the indication would fail downscale without detector voltage, but plausible because the SRM would need to be bypassed to continue the startup. If the candidate does not know that a rod block would come in then he may think that the startup can continue without bypassing the SRM.

C. Incorrect because the indication will not fail upscale, but plausible if the candidate does not know which direction the SRMs fail with a loss of detector voltage.

D. Incorrect because the startup can continue with one inoperable SRM per tech spec 3.1.1.b.4, but plausible if the candidate does not know the SRM operability requirements for control rod withdrawal.

References: N1-OP-38A

Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 40

The plant is operating at 87% power with the following conditions:

- APRM 16 is inoperable and bypassed for gain adjustments
- The red light on the Flow Control Trip Reference (FCTR) card for APRM 12 has stopped blinking (lit solid red)
- The ATC places APRM 12 joystick to bypass
- APRM 18 fails upscale due to electrical noise

Which one of the following describes the plant response?

A. A full scram will occur because a ½ scram exists from APRM 12 and APRM 18

B. A ½ scram on RPS channel 12 because APRM 18 is failed upscale and a Rod Block because APRM 12 is inoperable

C. A ½ scram only on RPS channel 12 because APRM 18 is failed upscale

D. A ½ scram and Rod Block on RPS channel 12 because APRM 18 is failed upscale

K&A # 215005 A3.07
Importance Rating 3.8

QUESTION 40

K&A Statement: A3.07 Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: RPS status

Justification:

A. Correct –APRM 18 failing high will give a ½ scram and APRM 12 is inop and the inop trip will have occurred. Bypassing 12 does not bypass 12 because 16 is already bypassed. The joy switch can be moved but an electrical interlock prevents 2 APRMs from the same quadrant being bypassed.

B. Incorrect because a full scram is the result of both RPS 11 and 12 having a half scram. Plausible if candidate thinks that the FCTR card will give a ROD Block.

C. Incorrect because a full scram is the result of both RPS 11 and 12 having a half scram. Plausible if candidate thinks that bypassing 12 will bypass the ½ scram for RPS 11.

D. Incorrect because a full scram is the result of both RPS 11 and 12 having a half scram. Plausible if candidate thinks bypassing 12 will bypass the ½ scram because the APRM 18 failing upscale will give a Rod Block and ½ scram.

References: N1-OP-38C Student Ref: None

Learning Objective: N/A

Question source: NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 41

- EDG 102 is carrying powerboard 102 while breaker R1012 is removed for the repair of an identified hot spot.

A fault on Battery Board #11 causes the battery board to trip and lock out.

No operator action has been taken. Which one of the following identifies the status of the ADS Logic Channels and ERV's that can actuate on an ADS signal?

	Available ADS Channels	ERV's Available for ADS
A.	Channel 11	Primary Valves
B.	Neither Channel	Primary Valves
C.	Channel 12	Backup Valves
D.	Both Channels	Backup Valves

K&A # 218000 K2.01
Importance Rating 3.1 (RO) / 4.3 (SRO)

QUESTION 41

K&A Statement: K2.01 – Knowledge of electrical power supplies to the following: ADS Logic

Justification:

A. A is incorrect – Power to the primary ADS valves comes from Battery Board #11. With the fault on BB 11 the Primary ADS Valves will only function in their mechanical pressure relief mode. Because of the alignment of Powerboard 102 with the diesel carrying the bus, when the DC battery board trips EDG 102 will coast to a stop and de-energize PB 102. Channel 11 of the ADS logic is de-energized by the loss of power because it is powered off of PB 102 and will not function due to the loss of the confirmatory logic. Therefore this answer is incorrect because both columns are incorrect. Plausible if the candidate does not know the which ADS valves are powered by Battery Board 11.

B. B is incorrect – Power to the primary ADS valves comes from Battery Board #11. With the fault on BB 11 the Primary ADS Valves will only function in their mechanical pressure relief mode. Because of the alignment of Powerboard 102 with the diesel carrying the bus, when the DC battery board trips EDG 102 will coast to a stop and de-energize PB 102. Channel 11 of the ADS logic is de-energized by the loss of power because it is powered off of PB 102 and will not function due to the loss of the confirmatory logic. Therefore this answer is incorrect because both columns are incorrect. Plausible if the candidate does not know the which ADS valves are powered by Battery Board 11

C. CORRECT C is correct. - Power to the backup ADS valves comes from Battery Board #12. With the fault on BB 11 the Backup ADS Valves are still available to function in their ADS mode. Because of the alignment of Powerboard 102 with the diesel carrying the bus, when the DC battery board trips EDG 102 will coast to a stop and de-energize PB 102. Channel 12 of the ADS logic is unaffected by the loss of power because it is powered off of PB 103 and will provide the signals required to open the backup ADS valves. This answer is correct.

D. D is incorrect – Power to the backup ADS valves comes from Battery Board #12. With the fault on BB 11 the Backup ADS Valves are still available to function in their ADS mode. Because of the alignment of Powerboard 102 with the diesel carrying the bus, when the DC battery board trips EDG 102 will coast to a stop and de-energize PB 102. Channel 11 of the ADS logic is de-energized by the loss of power because it is powered off of PB 102 and will not function due to the loss of the confirmatory logic. This answer is incorrect because Channel 11 is not available in this scenario. Plausible if the candidate does not know that channel 11 ADS logic has lost power.

References: NER-1M-095 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 2004 NRC

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 42

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

- RPS Bus 11 has tripped due to undervoltage relay failure
- N1-SOP-40.1, "Loss of RPS" has been entered

Describe the expected conditions for the RWCU system and its potential to be used as a pressure control system in the execution of EOP-1.

A. RWCU system isolates due to de-energizing the "valve control relays (4-11's)". RWCU can not be used for pressure control per EOP-1 Attachment 9 "RPV Press Control thru RWCU Temperature.

B. RWCU system isolates due to de-energizing the "valve control relays (4-11's)". RWCU can be used for pressure control per EOP-1 Attachment 9 "RPV Press Control thru RWCU Temperature.

C. RWCU system does not isolate until RPS Channel 12 trips and the "valve control relays (4-11's)" are de-energized. RWCU can be used for pressure control per EOP-1 Attachment 9 "RPV Press Control thru RWCU Temperature.

D. RWCU system does not isolate until RPS Channel 12 trips and the "valve control relays (4-11's)" are de-energized. RWCU can be used for pressure control per EOP-1 Attachment 9 "RPV Press Control thru RWCU Temperature or Attachment 8 "Attachment 8 "RPV Pressure Control thru Cleanup System Reject".

K&A #	223002 A2.03
Importance Rating	3.0 (RO) / 3.3 (SRO)

QUESTION 42

K&A Statement: A2.03 – Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System Logic Failures

Justification:

A. CORRECT A is correct – The RWCU system will isolate on a trip of RPS Channel 11 due to the de-energizing of RPS Bus #11 Ckt. #2 which will allow relays 4-11A, 11B and 11C to de-energize and provide isolation signals to the RWCU pumps and valves 33-02R, 33-01R, 33-04, and 33-41. De-energizing the 4-11's via circuit 2 disables the use of the EOP jumper from EOP-1 Attachment 9 and prevents the use of RWCU as an alternate pressure control system. Therefore Answer A is correct.

B. B is incorrect – As described above, the RWCU system will isolate due to the trip of the RPS bus and RWCU is not an available pressure control system even with the use of the EOP jumper specified in EOP-1 Attachment 9. Therefore answer B is incorrect. Plausible if the candidate believes that the EOP jumper will allow use of cleanup.

C. C is incorrect. - As described RWCU will isolate on the loss of power and does not require a trip signal from RPS Channel 12 to complete the isolation. As described previously RWCU is not available for pressure control per the EOP's with the loss of power. Plausible if candidate thinks it takes both channels of RPS to trip cleanup.

D. D is incorrect – As described RWCU will isolate on the loss of power and does not require a trip signal from RPS Channel 12 to complete the isolation. As described previously RWCU is not available for pressure control per EOP-1 Attachment 9 with the loss of power. Additionally, Attachment 8 of EOP-1 does not specify or allow the use of a jumper to bypass isolation signals to the RWCU valves to facilitate it's performance. Therefore Attachment 8 is also not available as an alternate pressure control operating mode with RPS Bus 11 tripped. Plausible if the candidate thinks that it takes both RPS channels to trip and that the EOPs are available.

References: N1-SOP-40.1
C-19859-C

Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 43

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

- Battery Board 12 is lost
- A sustained loss of Turbine Building ventilation is occurring
- Outdoor Temperature is 89 F

No operator actions are taken.

Which one of the following describes the anticipated plant response

A.	Only Rx Safety Valves open	1375 psig Safety Limit exceeded
B.	ERV and Safety Valves open	1375 psig Safety Limit not exceeded
C.	Only ERVs Lift	1375 psig Safety Limit exceeded
D.	No ERV and Safety Valves open	1375 psig Safety Limit not exceeded

K&A # 239002 K3.02
Importance Rating 4.2 (RO) / 4.4 (SRO)

QUESTION 43

K&A Statement: K3.02 – Knowledge of the effect that a loss or malfunction of the RELIEF / SAFETY VALVES will have on the following: Reactor over pressurization

Justification:

A. A is incorrect – With the loss of Turbine Ventilation, operation at full rated power and high outside air temperature, the initial conditions will eventually result in the closure of the MSIVs on high steam tunnel temperature. This will result in a high pressure and flux scram of the reactor. With the loss of Battery Board 12 the three backup ERV's will not function in their pressure relief mode. The primary ERV's will continue function in their pressure relief mode. Additionally, the high pressure safety limit is protected by the functioning of all the safety relief valves. Per Tech Spec 2.2.2 bases the safety limit will be maintained under these conditions with the operation of all nine relief valves. Therefore answer A is incorrect because it does not identify the opening of the three ERV's as part of the pressure transient or the protection from the safety limit being exceeded from the safety relief valves. This is a plausible answer if the bases for the pressure limit is not understood to include the Rx Safety Valves only.

B. CORRECT B is correct – As described above the initial conditions will result in a high pressure and flux scram of the reactor from the closure of the MSIV's. The primary ERV's will function in their pressure relief mode and all required reactor safety relief valves will operate. Per Tech Spec 2.2.2 bases the safety limit will be maintained under these conditions with the operation of all nine relief valves. Therefore answer B is correct because it identifies the opening of the three ERV's as part of the pressure transient and the protection from the safety limit being exceeded from the safety relief valves.

C. C is incorrect. - As described above both the ERV's and safeties are expected to lift in order to protect the pressure safety limit. This answer is incorrect due to it incorrectly identifying only the ERV's opening resulting in the pressure safety limit being exceeded. This answer is plausible if the candidate does not understand that a full power scram with MSIV closure will result in Rx Safety valve opening.

D. D is incorrect – As described above both the ERV's and safeties are expected to lift in order to protect the pressure safety limit. This answer is incorrect due to it incorrectly identifying that neither ERV's nor safety valves would open to maintain the pressure safety limit. This answer is plausible if the candidate does not recognize that the temperature conditions in the turbine building will result in closure of the MSIV's.

References: Tech Spec 2.2.2 BASES Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 2004 NRC

Question History: 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 44

The plant is performing a routine shutdown in preparation for refueling. A normal cooldown is being performed.

Which one of the following describes the desired level control strategy and adverse effect being avoided.

A. Control level on GEMAC level instruments (LI-36-76A & LI-36-77A) and account for uncompensated indicated level swell on Yarway level instruments (LI-36-09 & LI-36-10) to avoid unwanted turbine trip signals and high level alarms.

B. Control level on GEMAC level instruments (LI-36-76A & LI-36-77A) and account for uncompensated indicated level shrink on Yarway level instruments (LI-36-09 & LI-36-10) to avoid unwanted low level trip signals and RPS actuations.

C. Control level on Yarway level instruments (LI-36-09 & LI-36-10) and account for uncompensated indicated level swell on the GEMAC level instruments (LI-36-76A & LI-36-77A) to avoid unwanted turbine trip signals and high level alarms.

D. Control level on Yarway level instruments (LI-36-09 & LI-36-10) and account for uncompensated indicated level swell on the GEMAC level instruments (LI-36-76A & LI-36-77A) to avoid avoid unwanted low level trip signals and RPS actuations.

K&A #	259002 K1.09
Importance Rating	2.9 (RO) / 3.0 (SRO)

QUESTION 44

K&A Statement: K1.09 – Knowledge of the physical connections and/or cause-effect relationship between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Psat / Tsat (compensation)

Justification:

A. CORRECT A is correct – The GEMAC Narrow Range Level Instruments (LI-36-76A & LI-36-77A) are pressure compensated and are the preferred indication when the reactor is not at normal full power temperature and pressure per N1-OP-43A, B and C. The Yarway level indicators are not compensated and will drift up in indicated level as reactor temperature and pressure are reduced. The artificially high level may result in unwanted high level alarms as well as turbine trip signals while maintaining level on the compensated level instrumentation.

B. B is incorrect – As described above, the GEMAC level indicators are the correct level instruments to use for level control during the reactor depressurization. However, the uncompensated Yarway level indications will indicate a higher than actual level during the reactor depressurization, not a lower level as indicated in this answer. Therefore B is incorrect. Plausible if the candidate thinks the Yarways will trend lower vice higher.

C. C is incorrect. - As described above the Yarway level indications are not pressure compensated and are therefore not the desired indications to be used during the depressurization of the reactor. The GEMAC level indications are pressure compensated and will not result in turbine trip or high level alarms due to indicated level rise during the cooldown. Therefore C is incorrect. Plausible if the candidate thinks that the Yarways are pressure compensated and will track during cooldown.

D. D is incorrect – As described above the Yarway level indications are not pressure compensated and are therefore not the desired indications to be used during the depressurization of the reactor. The GEMAC level indications are pressure compensated and will not result in RPS trip signals being generated on low level due to indicated level drop during the cooldown. Therefore D is incorrect. Plausible if the candidate thinks that the Yarways are pressure compensated and will track during cooldown

References: N1-OP-43C Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 45

The plant is operating at 100% power with Reactor Building Ventilation in service when the following occurs:

- Alarm L1-4-3, REACT BLDG VENT RAD MONITOR OFF NORMAL is lit
- Reactor Building Exhaust Rad. Monitor 12 RN07B-5 low light is lit

Five minutes later the following occurs:

- RPS Bus 11 Normal Supply from UPS 162 Feeder breaker trips open

No operator actions have been taken.

Which one of the following describes the response of the RBEVS system given the current plant conditions?

A. Neither Reactor Building Rad monitor generate a trip signal and RBEVS does NOT start.

B. Reactor Building Exhaust Rad. Monitor 11 RN075A-5 generates a trip signal and RBEVS will start.

C. Reactor Building Exhaust Rad. Monitor 12 RN075B-5 generates a trip signal and RBEVS will start.

D. Both Reactor Building Rad Monitors generate a trip signal and RBEVS starts.

K&A # 261000 K6.04
Importance Rating 2.9 (RO) / 3.1 (SRO)

QUESTION 45

K&A Statement: K6.04 – Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Process radiation monitoring

Justification:

A. A is incorrect – The low/downscale reading of Reactor Building Exhaust Rad. Monitor 12 RN07B-5 will not produce a trip condition for a RBEVS system start. The loss of power due to de-energizing RPS Bus 11 will cause Reactor Building Exhaust Rad. Monitor 11 RN07A-5 to trip and provide a start signal to the RBEVs system. RBEVS will start from the single radiation channel trip. This answer is plausible since the intial low rad level does not produce a trip signal and the candidate must recognize the impact of the RPS breaker trip. If this is misunderstood or the single channel trip logic is not understood this answer will be selected.

B. CORRECT B is correct – The start of the RBEVS system will be initiated by the trip of the Reactor Building Exhaust Rad. Monitor 11 RN07A-5. Reactor Building Exhaust Rad. Monitor 12 RN07B-5 will not trip due to the downscale condition on the radiation montor.

C. C is incorrect. - The low/downscale reading of Reactor Building Exhaust Rad. Monitor 12 RN07B-5 will not produce a trip condition for a RBEVS system start. The loss of power due to de-energizing RPS Bus 11 will cause Reactor Building Exhaust Rad. Monitor 11 RN07A-5 to trip and provide a start signal to the RBEVs system. RBEVS will start from the single radiation channel trip. Therefore C is incorrect because the wrong radiation monitor is selected as the initiating signal to the trip. This is a valid distractor with the initial off normal alarm which is also used for the high level trip alarm. If the candidate does not understand the low trip does not cause an isolation then C will be selected.

D. D is incorrect – The low/downscale reading of Reactor Building Exhaust Rad. Monitor 12 RN07B-5 will not produce a trip condition for a RBEVS system start. The loss of power due to de-energizing RPS Bus 11 will cause Reactor Building Exhaust Rad. Monitor 11 RN07A-5 to trip and provide a start signal to the RBEVs system. RBEVS will start from the single radiation channel trip. Therefore D is incorrect because the wrong radiation monitors are selected as the initiating signals to the trip. This is a valid distractor based on the same reasons provided in answer C plus the correct answer from answer B.

References: N1-OP-10 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 46

The plant is operating at 100% power and the generator is supplying 100 MVAR:

Going to the "RAISE" position on the voltage regulator will cause the following parameter to change:

A. Generator speed to increase

B. Power factor to decrease

C. Field voltage to increase

D. Generator Output Voltage to increase

K&A #	262001 A1.03
Importance Rating	2.9 (RO) / 3.1 (SRO)

QUESTION 46

K&A Statement: A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: Bus voltage

Justification:

A. A is incorrect – Generator (turbine) speed in regulated by the turbine control valves in response to generator load. Turbine speed in maintained at 1800 rpm in order to maintain the frequency of the generator in synch with grid frequency. Changing voltage regulation will have no effect on turbine speed.

B. B is incorrect – The initial conditions place the generator supplying VARs to the grid which is a leading power factor. From these conditions raising field voltage to the generator will cause KVAR to increase and real power (MWe) to remain the same. This will cause power factor to increase not decrease as the answer states. If the generator had been in a lagging power factor this answer would have been correct.

C. CORRECT C is correct. - Raising generator field voltage increases field excitation and produces more MVARs from the generator. This is the intended purpose of taking the voltage regulator to "RAISE".

D. D is incorrect – The generator terminal voltage is unaffected by changes in voltage regulation with the generator synchronized to the grid. In order to raise terminal voltage increased power to the turbine must be provided from the reactor and the output breakers must be open to allow the voltage to move separately from the grid.

References: N1-OP-32 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.5

QUESTION 47

- The plant is operating at rated power in normal electrical alignment. During routine monthly surveillance testing, PB 102 Emergency Bus Under voltage relay 27-1X fails its surveillance and is placed in the tripped condition to support the Tech Spec 3.6.2i required action.

When the relay is tripped the following alarms are received:

- A4 1-6 POWER BD 102 BUS VOLTAGE LOW
- F138 PB 102 BUS VOLTS LOW LOW

While repairs are being started for the 27-1X relay, the following computer alarm is received:

- D199 PB 102 BUS VOLTS LOW LOW

- A technician supporting the repair verifies that the alarm is from PB 102 Emergency Bus degraded voltage relay 27-1AX.

- All indications of bus voltage are normal in the main control room.

What is the anticipated response of EDG 102 from these events?

A. Remains in standby.

B. Fast starts due to trip of bus undervoltage relay 27-1X and the output breaker remains open

C. Fast starts due to trip of bus degraded voltage relay 27-1AX and the output breaker closes

D. Fast starts due to trip of two relays 27-1X and 27-1AX and the output breaker closes.

K&A # 262001 K3.02
Importance Rating 3.8 (RO) / 4.2 (SRO)

QUESTION 47

K&A Statement: K3.02 – Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on the following: Emergency Generators

Justification:

A. CORRECT A is correct – The safeguard bus undervoltage and degraded voltage instrumentation is arranged in separate 2 out of 3 logic trains. For the conditions provided only one out the three undervoltage or degraded voltage relays tripped which will not cause a start of the diesel or allow it to close in and carry the effected power board. The diesel will remain in standby and the required Tech Spec actions for 3.6.2i will need to be taken for the degraded voltage relay 27-1AX similar to the actions taken for 27-1X.

B. B is incorrect – The trip of a single bus undervoltage relay will not cause the EDG to start. The output breaker will remain open because the diesel is not running and bus voltage is not actually degraded. Plausible if the candidate does not know that it takes 2 of 3 undervoltage relays to fast start the EDG.

C. C is incorrect. - The trip of a single bus degraded voltage relay will not cause the EDG to start. The output breaker will remain open because the diesel is not running and bus voltage is not actually degraded. Plausible if the candidate does not know that it takes 2 of 3 undervoltage relays to fast start the EDG

D. D is incorrect – The trip of a single bus undervoltage relay combined with a single bus degraded voltage relay will not cause the EDG to start. Each of the diesel generator initiation circuits on bus voltage are independent of one another and the combination of one in each logic trip will not produce a diesel start. The output breaker will remain open because the diesel is not running and bus voltage is not actually degraded. Plausible if the candidate does not know that there are two separate circuits for loss of voltage and degraded voltage.

References: N1-OP-32 Student Ref: None

Learning Objective: N/A

Question source: Modified OC 2006 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 48

- The plant is operating at 100% power with all equipment in its normal electrical alignment. UPS 162A is in service when its inverter fails and the inverter output goes to zero.
- Two minutes later the common supply breaker for Static Battery Chargers 161A and 161B trips open.

No operator actions have been taken.

Which one of the following describes the effect on RPS Bus 11 and the source from which it is being powered, if any?

A. Powered by UPS 162B with no loss of power.

B. Powered by UPS 162A with 125VDC from Battery Board 11 with no loss of power.

C. Power is lost and must be manually transferred to I&C Bus 130A

D. Powered by UPS 162A with 600 VAC from Power Board 16 Section B with no loss of power.

K&A # 262002 K4.01
Importance Rating 3.1 (RO) / 4.2 (SRO)

QUESTION 48

K&A Statement: K3.02 – 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

Justification:

A. A is incorrect – With an inverter failure the UPS will automatically swap to the bypass power supply step down transformer. If an additional failure of the AC power supply occurred the load would be transferred to the alternate DC supply which does not use the inverter. The power will not automatically transfer to the 162B inverter. The transfer between inverters requires the use a manual transfer switch to allow the alternate UPS to provide power to the effected loads. Therefore A is incorrect. Plausible because the UPS 162B could be lined up to power these loads however this requires a manual transfer.

B. B is incorrect – The inverter failure will cause the loads to transfer to the bypass power supply step down transformer. A failure of the AC input into the UPS or a failure of the UPS rectifier will cause the loads to shift to the 125 VDC source. The DC source is still available in this scenario with only the battery chargers being offline and the DC feed remaining energized by the battery cells. Therefore there is no transfer to the DC feed of the UPS and this answer is incorrect. Plausible if the candidate does not know that the UPS will not transfer to the DC supply in this set of plant conditions.

C. C is incorrect. - The inverter failure will cause the loads to transfer to the bypass power supply step down transformer. Because powerboard 16B continues to be energized and powering RPS Bus 11 there is no loss of power and no need to transfer the power supply to I&C Bus 130A. Therefore answer C is incorrect. Plausible if the candidate does not know that the inverter failure will cause a transfer to the bypass power supply without a loss of load.

D. CORRECT D is correct – With an inverter failure the UPS will automatically swap to the bypass power supply step down transformer. The loads downstream of UPS 162A will continue to be energized through the step down transformer until it is removed from service for repairs. The failure of the supply breaker into the DC battery board will have no effect on the UPS because the battery board will remain energized from the #11 battery. Section B of powerboard 16 is still the source of power to RPS Bus 11.

References: N1-OP-40, C-19436-C Sheets 1 & 6 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.7

QUESTION 49

The plant is operating at 100% power with all equipment in its normal electrical alignment and SBC 172A in service. The following alarm is received:

- A3 4-4 BAT. BD. 12 BATTERY BREAKER TRIP

No operator actions have been taken.

Which one of the following describes the extent of the DC power loss, if any?

A. Charger 172A maintains DC power to Battery Board 12 without a loss of power

B. DC Valve board 12 transfers to Battery Board 11. All other DC power from Battery Board 12 is lost.

C. MG Set 167 swaps to “BATTERY CHARGE” mode and supplies battery board 12 after a loss of power.

D. A complete loss of DC power to Battery Board 12 occurs

K&A # 263000 K1.02
Importance Rating 3.2 (RO) / 3.3 (SRO)

QUESTION 49

K&A Statement: K1.02 – Knowledge of the physical connections and/or cause-effect relationships between DC ELECTRICAL DISTRIBUTION SYSTEM and the following: Battery charger and battery

Justification:

A. A is incorrect –The alarm that was received identifies that the #12 Battery Breaker has tripped. With a normal electrical alignment the trip of the battery breaker will result in a trip of the battery charger DC output breaker as well due to their electrical interlock. The distracter is plausible if the applicant forgets the interlock. The trip of the common DC output breaker will not allow SBC 172A to supply BB 12. Therefore A is incorrect.

B. B is incorrect – All DC power from BB12 including the loads on DC Valve Board 12 will be lost due to the trip of the #12 Battery Breaker and subsequent trip of the SBC DC breaker. The distracter is plausible because the loads on DC Valve board 12 can be manually realigned to Battery Board 11 but no auto transfer will occur to allow that cross-tie. Therefore B is incorrect.

C. C is incorrect. - MG Set 167 is capable of supplying BB 12 if it were to be de-energized. The MG Set would be placed in its "BATTERY CHARGE" mode of operation and aligned to BB 12 to support operation of the components fed from the battery board. The distracter is plausible if the applicant does not remember that the transfer of MG Set 167 to its "battery charge" mode of operation and alignment to BB12 must be done manually in the field vice an automatic operation. Therefore answer C is incorrect.

D. CORRECT D is correct – With the trip of #12 Battery Breaker the normal electrical alignment will produce a of the battery charger DC output breaker as well due to their electrical interlock. The trip of the common DC output breaker will result in a total loss of DC to Battery Board 12. There are no automatic features enabled to repower the battery board or the panels it feeds. Therefore D is correct.

References: N1-OP-47A, N1-ARP-A3 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: 2005 Retake Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 50

Given the following conditions for Diesel Generator (DG) 103:

- A leak in the common header downstream of the DG 103 Air Compressors occurs
- Air Receiver Tank pressures are 200# and lowering
- THEN an automatic start signal is received
- 5 seconds after start signal the engine speed reaches 175 rpm
- Annunciator A4-3-5 DIESEL GENERATOR START-RUN OFF NORMAL alarms

Which one of the following describes the automatic response of the DG and the manual operator actions required to attempt a restart of the DG?

A. The DG attempts a second start. If that start fails, the operator must correct the start failure condition and place the engine control switch to EM STOP.

B. The DG attempts a second start. If that start fails, the operator must correct the start failure condition and depress the RESET / FAST STOP and the incomplete sequence circuit (48x) pushbuttons.

C. The DG shuts down immediately and locks out. The operator must correct the start failure condition and depress the RESET / FAST STOP and the incomplete sequence circuit (48x) pushbuttons.

D. The DG shuts down immediately and locks out. The operator must correct the start failure condition and place the engine control switch to EM STOP.

K&A # 264000 A3.01
Importance Rating 3.0 (RO) / 3.1 (SRO)

QUESTION 50

K&A Statement: A3.01 – Ability to monitor automatic operations of the EMERGENCY DIESEL GENERATORS including: Automatic starting of compressor and emergency generator

Justification:

A. A is incorrect – In order to allow subsequent starts of the diesel following a second failed attempt the incomplete sequence circuit and fast stop circuits must be reset. These can only be reset by pressing the pushbuttons in the local panel. Use of the EM Stop position on the diesel control switch energizes the 5DE fast stop and reset relay. This stops the diesel but does not reset the Fast Stop or Incomplete sequence circuits. Therefore A is incorrect. Plausible if the candidate does not know that they need to locally reset the EDG to attempt another start.

B. CORECT B is correct – If the diesel engine does not attain 200 rpm in five seconds the diesel fast stops. The diesel then attempts a second start. If the diesel does not attain 750 rpm in 2 minutes the diesel will shutdown and the incomplete sequence circuit (48x) and RESET / FAST STOP pushbuttons must be reset to attempt another start of the diesel

C. C is incorrect - The diesel will attempt a second start prior to requiring manual action form the operator. Although the actions in this answer are correct, they are not required prior to the second automatic start occurring. Plausible if the candidate does not know that a second start will be attempted automatically.

D. D is incorrect – The diesel will attempt a second start prior to requiring manual action form the operator. Additionally the actions in this answer are incorrect as described in Answer A., and are not required prior to the second automatic start occurring. Plausible if the candidate does not know that a second start will be attempted automatically

References: 1101-264001C01 EDG instructors Guide   Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: 2005 NRC Retake Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 51

103 Diesel Generator is operating in parallel with Power Board 103 per N1-ST-M4B with the following indications:

- 60 Hz
- 500Kw
- 60 KVAR

WHICH ONE of the following actions will establish its correct operating limits?

A. Place DIESEL GOV control switch in RAISE to raise leading VARs.

B. Place DIESEL GOV control switch in RAISE to raise lagging VARs.

C. Place VOLT ADJ RHEO GEN 103 in RAISE to raise lagging VARs.

D. Place VOLT ADJ RHEO GEN 103 in RAISE to raise leading VARs.

K&A # 264000 A4.05
Importance Rating 3.6 (RO) / 3.7 (SRO)

QUESTION 51

K&A Statement: **A4.05-** Ability to manually operate and/or monitor in the control room: **Transfer of emergency generator (with load) to grid.**

Justification:

A. A is incorrect – This is the wrong control switch and it is used for speed control. The increase in diesel speed will directly impact real power (kW) output and have a minor effect on reactive loading. This is a valid distractor because it presents the only other switch to be manipulated during the performance of the monthly diesel operability surveillance and the candidate must understand the power factor that the diesel normally operates with.

B. B is incorrect – This is the wrong control switch and it is used for speed control. The increase in diesel speed will directly impact real power (kW) output and have a minor effect on reactive loading. This is a valid distractor because it presents the only other switch to be manipulated during the performance of the monthly diesel operability surveillance and the candidate must understand the power factor that the diesel normally operates with.

C. CORRECT C is correct - Per N1-ST-M4B Step 8.1.16, "If performing N1-ST-M4A alone adjust VOLT ADJ RHEO GEN 103 switch to establish reactive load between 300 and 800 KVARS". The switch must be placed in RAISE to pick up lagging vars. The candidate must recognize that the voltage rheostat can be adjusted with the output breaker closed and that the diesel operates in a lagging power factor (supplying vars to the grid) during parallel operation with the grid.

D. D is incorrect – Placing the voltage rheostat to RAISE will increase the lagging VARs the diesel is carrying. This is a plausible distractor because the candidate must recognize the correct switch to be used and understand the power factor that the diesel normally operates with.

References: 1101-264001C01 EDG instructors Guide   Student Ref: None

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: 2005 NRC Retake Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 52

The plant is at 100% power with the following:

Instrument Air Compressor (IAC) #13 in service
IAC #12 in pull-to-lock
Instrument Air Dryer (IAD) 94-168 in service
Subsequently, IAC #11 trips (lost control power) and its control switch is placed in pull-to-lock
The ASSS directs bypass of IAD 94-168 and IAD 94-169

Per N1-OP-20, which one of the following actions is required until either IAC #11 or IAC #12 is returned to service?

A. Blow down designated air manifolds once every 24 hours.

B. Align the temporary service air compressor to the service air system.

C. Align service air to the reactor building track bay roll door (D-39) inflatable seal.

D. Align breathing air to the Service Air System after removing the valve internals from check valve 94-51.

K&A #	300000 A2.01
Importance Rating	2.9 (RO) / 2.8 (SRO)

QUESTION 52

K&A Statement: **A2.01 –** Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control , or mitigate the consequences of those abnormal operation: **Air dryer and filter malfunctions**

Justification:

A. CORRECT A is correct – Per N1-OP-20; D.4.0: Any combination of pulling control power fuses AND/OR placing Control Switches to Pull To Lock for both Instrument Air Compressors 11 AND 12 will result in a Loss of Control Power to the Instrument Air Dryers 94-168 and 94-169, resulting in a shutdown of the Instrument Air Dryers. Per N1-OP-20, Section H.3.0, step 3.6, blow down designated air manifolds daily until air dryers are restored.

B. B is incorrect – Aligning the temporary service air compressor to service air will not provide additional back-up to supply instrument air. With both IACs in PTL the service air compressor will be supplying instrument air through BV 94-19. When shutting down both IACs IAW N1-OP-20 H.17.9 verifies that the temporary service air compressor is NOT supplying Service Air. This is a plausible distractor because the use of the temporary Service Air compressor is referenced in the section detailing the shutdown of IACs 11 and 12.

C. C is incorrect - Per N1-OP-20; H.17.0 note: The Reactor Building Track Bay Roll Door (D-39) inflatable seal is pressurized from the Instrument Air System and cannot be considered operable with IAC 11 and IAC 12 removed from service. Therefore, failure to maintain the Rx Bldg Outer Swing Door (D-198) closed and sealed while IAC 11 and IAC 12 are removed from service will result in violation of secondary containment integrity. Per N1-OP-20; H.17.0, Step 17.4: If secondary containment integrity is required, THEN verify the following: (1) Rx Bldg Outer Swing Door (D-198) is closed and sealed, and (2) Clearance section placed on Rx Bldg Outer Swing Door (D-198) in the closed and sealed position. This is a plausible distractor because the operability of the track bay roll door seal is impacted by securing IACs 11 and 12, however the required action is incorrect.

D. D is incorrect – The check valve internals from 94-51 must be removed to align IAC 13 to service breathing air. This is a plausible distractor because the removal of the check valve internals and the realignment of breathing air are both described in Section 17 of N1-OP-20, however the source of the air is incorrect.

References: N1-OP-20 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC Exam

Question History: NMP1 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 53

The plant is operating with the following lineup:

- TBCLC is in a normal lineup
- 11 TBCLC pump is running
- 12 and 13 TBCLC heat exchangers are in service
- Condenser intake temperature is 45F
- A loss of instrument air to TBCLC temperature controller TC-71-20 occurs

Which one of the following is the expected change in the TBCLC supply temperature and the action to be taken to control this temperature?

	Temperature Response	Action to Control Temperature
A.	Rise	Throttle SW Drag valve bypass 72-93R BYPASS AROUND TURB BLDG SW DRAG VALVE
B.	Rise	Manually control TCV 71-88 TBCLC TCV and TCV 72-147 SW TCV
C.	Lower	Throttle SW Drag valve bypass 72-93R BYPASS AROUND TURB BLDG SW DRAG VALVE
D.	Lower	Manually control TCV 71-88 TBCLC TCV and TCV 72-147 SW TCV

K&A #	400000 K6.03
Importance Rating	2.9 (RO) / 3.0 (SRO)

QUESTION 53

K&A Statement: **K6.03 –** Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: **Controllers and Positioners**

Justification:

A. A is incorrect – On a loss of control air signal from E/P 71-20, HX TCV 71-88 will fail as is and SW TCV 72-147 will fully open fully maximizing cooling. With SW flow maximized heat exchanger outlet temperature will lower. Opening of the SW drag valve will continue to lower temperature by increasing SW flow through the heat exchangers. This is a plausible distracter because use of the SW drag valve is an alternative temperature control mechanism which is proceduralized in N1-OP-24 and for candidates that believe the TBCLC and/or SW TCV close on a loss of air and therefore the TBCLC temperature will rise.

B. B is incorrect – On a loss of control air signal from E/P 71-20, HX TCV 71-88 will fail as is and SW TCV 72-147 will fully open fully maximizing cooling. With SW flow maximized heat exchanger outlet temperature will lower. This is a plausible distracter for candidates that believe the TBCLC and/or SW TCV close on a loss of air and therefore the TBCLC temperature will rise.

C. C is incorrect - On a loss of control air signal from E/P 71-20, HX TCV 71-88 will fail as is and SW TCV 72-147 will fully open fully maximizing cooling. Opening of the SW drag valve will continue to lower temperature by increasing SW flow through the heat exchangers. This is a plausible distracter because use of the SW drag valve is an alternative temperature control mechanism which is proceduralized in N1-OP-24.

D. CORRECT D is correct – On a loss of control air signal from E/P 71-20, HX TCV 71-88 will fail as is and SW TCV 72-147 will fully open fully maximizing cooling. With SW flow maximized at this low a lake temperature; heat exchanger outlet temperature will lower. To raise the heat exchanger outlet temperature, take manual control of SW TCV and/or HX TCV 71-88 and operate the valves to raise temperature.

References: N1-OP-20, Section H.9 and H.11 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2005 NRC Exam

Question History: NMP1 2005 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 54

- The plant is operating at 100% power with both the Reactor Building Inner Lift Door and the Outer Swing Door closed.

- An operator is tasked with opening the Outer Swing Door.
- Prior to opening the Outer Swing Door he verifies closed the Inner Lift Door, and notes that the Seal Pressure is 3 psig.

What is the impact on Secondary Containment and what are the required actions?

A. Secondary Containment is not met and Technical Specification 3.4.1, Secondary Containment Leakage Rate requires restoring the leak rate to within specified limits within 4 hours.

B. Secondary Containment is met and the Outer Swing Door can be opened per N1-OP-52.

C. Secondary Containment is not met and RBEVS must be started to maintain the Reactor Building at a negative pressure.

D. Secondary Containment is met but the Outer Swing Door can not be opened.

K&A # 290001 A2.01
Importance Rating 3.3

QUESTION 54

K&A Statement: Ability to (a) predict the impacts of the following on the Secondary Containment; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Personnel airlock failure.

Justification: Exam Team analyzed question and K/A. The team feels this is a match because the question tests RO system level knowledge of the impact on secondary containment and the interlock at a higher cognitive level, without going into the SRO level by controlling the consequences through use of technical specifications.

A. Incorrect. Secondary containment is met with the outer door shut. Plausible if the applicant thinks that the inner door not being sealed makes secondary containment inop. Which would require entering the LCO.

B. Incorrect. Plausible if the applicant does not think that the inner seal being at 3 psig would make the door inop and therefore can not open the door.

C. Incorrect. Secondary containment is met with the outer door shut. Plausible if the applicant thinks that the inner door not being sealed makes secondary containment inop.

D. Correct – Secondary Containemnt is met, however with the seal below 7 psig, the outer door is interlocked and will not open due to the inner door may be inoperable with the seal below 5 psig.

References: N1-OP-52 Student Ref: None

Learning Objective: N/A

Question source: NMP1 modified

Question History: 2004 NRC exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5

QUESTION 55

The plant is planning for an upcoming refueling outage, with the following:

- Work on Recirc Loop 15 will require plugging both the suction and discharge nozzles
- All other Recirc Loops will be available

Which one of the following describes Shutdown Cooling availability and the reason?

Shutdown Cooling would be...

A. Unaffected due to Recirc Loop 14 being available.

B. Unavailable due to the loss of the only system suction path.

C. Unavailable due to the loss of the only system discharge path.

D. Unavailable due to the loss of both the system suction and discharge paths.

K&A # 202001 K1.18
Importance Rating 3.3

QUESTION 55

K&A Statement: Knowledge of the physical connections and/or cause and effect relationships between RECIRCULATION SYSTEM and the following: RHR shutdown cooling mode.

Justification:

A. Incorrect because the SDC system does not have a complete path for flow with the 14 Recirc loop.

B. Incorrect because discharge path and not the suction path is lost.

C. Correct – The SDC system takes a suction on Recirc Loop 14 and returns via Recirc Loop 15. The system discharge path will be lost.

D. Incorrect because only the discharge path is lost.

References: N1-OP-04 B.2 page 4 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2002

Question History: NRC Exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.2-
41.9

Comments:
TRH 7/17/09 – Minor editorial changes to stem.

QUESTION 56

A Reactor Startup is in progress, with the following:

- RWM is operable
- Reactor Power is 4%
- Group 8 rods are latched
- Group 7 Insert and Withdraw limits are 04 and 12 respectively
- Group 8 Insert and Withdraw limits are 12 and 24 respectively
- All Group 7 Rods except two, are at 12
- The remaining Group 7 rods are at 04
- Group 8 Rods are at their Insert Limit position 12

A Group 8 Rod is selected with the intention to withdraw the rod to position 24. The operator goes the wrong direction and inadvertently inserts the rod.

What is the response of the rod and at what point, if any, will a RWM rod block will be generated?

A. Rod will not move, a select error Rod Block is generated when the rod is selected.

B. Rod will insert to 10 and an insert error Rod Block will be generated.

C. Rod will insert to 00. No Rod Blocks are generated.

D. Rod will insert to 08 and an insert error Rod Block will be generated.

K&A # 201006 K5.11
Importance Rating 3.2

QUESTION 56

K&A Statement: Knowledge of the operational implications of the following concepts as they apply to ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) : **Insert error.**

Justification:

A. Incorrect. Plausible if the applicant thinks that a select error will be generated. A select error is generated when a rod is selected that is not in the currently latched group. The rod selected is in the latched group.

B. Incorrect. Plausible if the applicant thinks the error will happen at 10 which is beyond the insert limit. However at 10 the alternate limit is satisfied and there will not be a rod block.

C. Incorrect. Plausible if the applicant does not realize that this rod is the 3rd insert error which will generate a rod block.

D. Correct – With 2 rods left in a group the RWM will move up to detecting the next group and therefore the 2 remaining rods will show an insert error, meaning there are rods in a lower group that are not withdrawn to their limit. With the 3rd rod giving an insert error there will be a Rod Block. The insert error will occur at the alternate limit which is one notch past the insert limit.

References: N1-OP-37 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.5

QUESTION 57

The reactor is at rated power.

Which of the following annunciators correspond to an EOP entry condition? (Assume the alarms are received individually and are valid.)

A. RPS CH 11 Reactor Press High

B. Rx BLDG Area Temp High

C. Torus Water Level High-Low

D. RPS CH 11 Reactor Level Low

K&A # 216000 G2.4.2
Importance Rating 4.5

QUESTION 57

K&A Statement: Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions as related to Nuclear Boiler Instrumentation.

Justification:

A. Incorrect because high pressure alarm is set at 1068 psig and the EOP entry condition is 1080 psig. Plausible if the candidate does not know the alarm is set below the EOP entry condition.

B. Correct – Reactor Building Area High Temperature alarm, if valid is an EOP entry condition for EOP-5.

C. Incorrect because Torus high level alarm is at 11.1 feet and EOP entry is 11.25 Feet. Torus Low Level alarm is 10.65 Feet and EOP entry is 10.5 Feet. Plausible if the candidate does not know the alarm is set above and below the EOP entry condition.

D. Incorrect because reactor level low alarm set at 55.5 inches and the EOP entry condition is 53 inches. Plausible if the candidate does not know the alarm is set above the EOP entry condition.

References: ARP's K1-1-1, F1-1-2, F1-1-3, K3-3-1 Student Ref: None

Learning Objective: N/A

Question source: Modified OC 2006

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 58

Plant Conditions are as follows:
100% Power

- Torus Temperature is 80 degrees and rising slowly
- 4160KV Power Board 102 trips on a ground fault.

Which one of the following describes the containment spray loops available for Torus Cooling per N1-OP-14 based on the above conditions?

A. Loops 111 and 112

B. None, because 80-118 Containment Spray Test To Torus FCV does not have electrical power.

C. Loops 121 and 122

D. Loops 112 and 122

K&A # 219000 K6.01
Importance Rating 3.2

QUESTION 58

K&A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI:Torus/Suppresion Pool Cooling Mode: A.C. Electrical Power.

Justification:

A. Incorrect but plausible if the applicant thinks the 111 and 112 are powered from PB 103, CS pumps and CS RW pumps 111 and 112 are powered from PB 102 and can not be used.

B. Incorrect but plausible, the normal supply to 80-118 is from PB 102 to PB167, however PB 167 will auto transfer to the alternate power source.

C. Correct – CS pumps and CS RW pumps 121 and 122 are powered from PB 103 and are unaffected by the loss of PB 102.

D. Incorrect but plausible, CS pump and CS RW pump 112 are powered from PB 102 and can not be used.

References: N1-OP-14 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.7

QUESTION 59

Which of the following is the normal makeup water flow path to the Spent Fuel Storage Pool Filtering and Cooling System to compensate for the lowering water level due to evaporative losses from the Spent Fuel Pool?

A. Surge tank level is sensed which opens a level control valve to admit water to the surge tank.

B. Fuel Pool level is sensed which opens a level control valve to admit water to the surge tank.

C. Surge tank level is sensed which opens a level control valve to admit water directly to the fuel pool.

D. Fuel Pool level is sensed which opens a level control valve to admit water to the directly to the fuel pool.

K&A # 233000 K4.06
Importance Rating 2.9

QUESTION 59

K&A Statement: Knowledge of FUEL POOL COOLING AND CLEAN-UP design feature(s) and/or interlocks which provide for the following: **Maintenance of adequate pool level.**

Justification:

A. Incorrect. Plausible because the backup valve 57-58 discharges to the surge tank.

B. Incorrect. Plausible because the backup valve 57-58 discharges to the surge tank and it is plausible that the fuel pool is directly sensed for level control.

C. Correct –Surge Tank Level is sensed by LT 54-27 and opens the normal makeup valve LCV 57-25, which discharges directly to the fuel pool.

D. Incorrect. Plausible that the fuel pool is directly sensed for level control.

References: N1-233000-RBO-03-Q-5 Student Ref: None

Learning Objective: N/A

Question source: NMP1 bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 60

Which one of the following describes the design bases closure time limit and the basis for the Main Steam Isolation Valve closure time limit?

	MSIV Closure Time Limit	Basis for MSIV Closure Time Limit
A.	3 Seconds	Valve closure time limit in conjunction with flow limiters ensure core coverage
B.	3 Seconds	Valve closure time in conjunction with limit switch based SCRAM ensures pressure transient does not require ERV discharge
C.	10 Seconds	Valve closure time limit in conjunction with flow limiters ensure core coverage
D.	10 Seconds	Valve closure time in conjunction with limit switch based scram ensures pressure transient does not require safety valve discharge

K&A #	239001 G2.1.28
Importance Rating	4.1 (RO) / 4.1 (SRO)

QUESTION 60

K&A Statement: **G2.1.28** – Knowledge of the purpose and function of major system components and controls.

Justification:

A. A is incorrect – The 3 second time limit for valve closure is part of the Chapter 15 analysis to limit the pressure transient from a Main Steam Line Isolation Valve closure (with SCRAM) to prevent opening safety valves and maintaining the pressure transient within the capability of the ERV's. The three second isolation time is NOT used as part of the basis for limiting inventory loss and maintaining core coverage during a MSL break accident. This is a plausible distractor because both the isolation time and bases are individually correct but are not associated with one another.

B. B is incorrect – The 3 second time limit for valve closure is part of the Chapter 15 analysis to limit the pressure transient from a Main Steam Line Isolation Valve closure (with SCRAM) to prevent opening safety valves and maintaining the pressure transient within the capability of the ERV's. This transient is expected to cause ERV actuations and is incorrect because of the reference to no ERV actuations in the bases column. This is a plausible distractor because the isolation time and bases are correct except for the use of safety valve versus ERV.

C. CORRECT C is correct – Per the UFSAR Section 15 Accident Analysis for the Main Steam Line Break Outside the drywell the assumptions of the accident analysis show that the combination of the flow limiters and isolation valve closure limits when offset by feedwater flow addition limits the loss of water volume to maintain water level above the core. This maintains the core coverage safety limit.

D. D is incorrect – The 10 second isolation time is associated with the limitations of coolant loss and not the limitations of the pressure transient due to MSIV closure. This is a plausible distractor because both the isolation time and bases are individually correct but are not associated with one another.

References: FSAR Section XV 3.5.1 & C.1.2 Student Ref: None

Learning Objective: N/A

Question source: NEW

Question History: NEW

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.7

QUESTION 61

Unit 1 is operating at 90% power with the following conditions:

- All Condensate pumps in service
- Four (4) Filter Vessels in service
- Six (6) Condensate demineralizers in service

Condensate Pump #13 must be removed from service due to scheduled electrical testing.

What are the required actions (if any) to allow removal of the #13 condensate pump from service.

A. No actions are required. Bypass valve AO-201 remains closed with the condensate pump out of service.

B. Bypass valve AO-201 shall be OPEN to assure HPCI operability with the condensate pump out of service.

C. Throttle open Bypass valve AO-201 to assure HPCI operability with the condensate pump out of service.

D. Throttle open Bypass valve AO-201 to maintain FW Booster suction pressure >65 psig with the condensate pump out of service.

K&A # 256000 A4.02
Importance Rating 2.8 (RO) / 2.8 (SRO)

QUESTION 61

K&A Statement: **A4.02-** Ability to manually operate and/or monitor in the control room: System motor operated valves

Justification:

A. A is incorrect – Actions are required to be taken to address securing the #13 condensate pump as described in answer D. This is a plausible distractor since several combinations of condensate pumps, filters and demins shown in Attachment 4 of N1-OP-15C do not require the bypass valve to be opened to support their removal from service.

B. B is incorrect – The full opening of the bypass valve is only required with the combination of 2 condensate pumps, 3 filter vessels and 5 demins in service. This is a plausible distractor since this action is presented in Attachment 4 of N1-OP-15C; however it is not required for the proposed evolution.

C. C is incorrect – With the conditions described in the stem (2 condensate pumps, 4 filter vessels and 6 demins in service) the required action is to throttle the bypass valve to maintain >65 psig at the suction of the FW booster pumps. This answer is incorrect because the action is not being taken to maintain HPCI operability. This is a plausible distractor because it combines the correct action with the improper basis, both of which are present in Attachment 4 of N1-OP-15C.

D. CORRECT D is correct – With the conditions described in the stem (2 condensate pumps, 4 filter vessels and 6 demins in service) the required action is to throttle the bypass valve to maintain >65 psig at the suction of the FW booster pumps. This provides pump protection and avoids spurious trips of the booster pumps in the off-normal alignment.

References: N1-OP-15C Student Ref: None

Learning Objective: N/A

Question source: NEW

Question History: NEW

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.7

QUESTION 62

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

- RWCU pump 12 is in service
- RWCU Surge Tank Level is reported to be at 66%

Which one of the following describes the **transient effect** of the actions required to be taken to correct the RWCU Surge Tank Level?

A. RWCU System Flow rises; Core Thermal Power rises

B. RWCU System Flow lowers; Core Thermal Power lowers

C. RWCU Suction Pressure rises; RWCU return to vessel temperature rises

D. RWCU Suction Pressure lowers; RWCU return to vessel temperature lowers

K&A #	204000 A4.06
Importance Rating	2.8 (RO) / 2.8 (SRO)

QUESTION 62

K&A Statement: **A4.06** – Ability to manually operate and/or monitor in the control room: System Flow

Justification:

A. A is incorrect – Increasing the suction pressure in the Surge tank does not improve system flow per the CAUTION prior to Step F.9.1. Even though the temperature in the system drops due to the reduced system flow, the change in system flow causes a more significant change in Core Thermal Power and causes power to drop versus rise due to the addition of colder water. This is a plausible distractor because the improved suction pressure would normally cause an improvement in pump and system performance and that improved flow would then lead to increased thermal power.

B. CORRECT B is correct – RWCU surge tank is required to be maintained between 40% - 50% per N1-OP-3 Section F.9. The surge tank level is reduced by increasing the nitrogen pressure in the tank and thereby increasing suction pressure. Per the CAUTION prior to Step F.9.1, this change in suction pressure will result in a drop in system flow and a resultant drop in CTP. The parameters that will change during the evolution are expected to return to normal after the pressurization is complete and the system stabilizes in its new condition.

C. C is incorrect –Suction pressure does increase due to this evolution however this does not result in improved system flow per the CAUTION prior to Step F.9.1. This is a plausible distractor because the improved suction pressure would normally cause an improvement in pump and system performance and that improved flow would then lead to increased temperature at the outlet of the REGEN and NON-REGEN Heat Exchangers and the RWCU return temperature to the vessel.

D. D is incorrect – Suction pressure does not decrease due to this evolution because inventory is not removed from the surge tank, pressure is increased in the tank. This is a plausible distractor for those candidates that do not understand the mechanism to reduce surge tank level. This would lead to a temporary reduction in suction pressure and reduced temperatures in the system due to reduced overall system flows.

References: N1-OP-3 Student Ref: None

Learning Objective: N/A

Question source: NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 63

An electrical malfunction causes a spurious actuation of the CO2 system for the Records Storage Room. Which one of the following describes how personnel exposure to carbon dioxide is prevented?

The horn and siren alarm for ___(1)___ before carbon dioxide discharge, and wintergreen oil is discharged ___(2)___ carbon dioxide.

	(1)	(2)
A.	30 seconds	with
B.	30 seconds	10 seconds before
C.	60 seconds	with
D.	60 seconds	30 seconds before

K&A # 286000 K4.04
Importance Rating 3.6 (RO) / 3.7 (SRO)

QUESTION 63

K&A Statement: **K4.04** - Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: **Personnel safety during halon and/or carbon dioxide system actuation**

Justification:

A. CORRECT A is correct – The Record Storage Room has a mechanical 30 second time delay device with a predischarge alarm horn and light located in the record storage room. The wintergreen capsule is located with the hazard selector valve. With the actuation of the system the alarm will sound for 30 seconds and the flow of CO2 will rupture the odorizing capsule when the system actuates.

B. B is incorrect – The odorizer in the CO2 system is carried along with the CO2 discharge. There is no means to have the wintergreen odor occur ahead of the system discharge. This is a plausible distracter for those candidates that believe the wintergreen odor is discharged as a preparatory action ahead of the system discharge.

C. C is incorrect – The mechanical timer is set for 30 seconds on the Records Storage Room. This is a plausible distracter for those candidates that are unsure of the timer design of the Record Storage Room or believe the timer is different from other areas of the plant which also use a 30 second time delay prior to system discharge.

D. D is incorrect – The odorizer in the CO2 system is carried along with the CO2 discharge. There is no means to have the wintergreen odor occur ahead of the system discharge. This is a plausible distracter for those candidates that believe the wintergreen odor is discharged as a preparatory action ahead of the system discharge.

References: N1-OP-21C Student Ref: None

Learning Objective: N/A

Question source: NMP1 Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.5

QUESTION 64

With the plant at 100% power, the containment spray system controls are aligned for standby operation per N1-OP-14. Subsequently, the following events occur:

- A seismic event occurs
- The seismic event triggers a large break LOCA occurs that provides valid automatic initiation signals to the Containment Spray System
- The Northeast Containment Spray Compartment is flooded above the Maximum Safe Level and the pumps in the room are not able to be used

Which one of the following is the design response of containment spray to provide rated spray flow in response to the above conditions?

A. Less than design bases spray flow because two 100% capacity pumps are operating in the same loop and can not meet the requirements for the Appendix J Containment Spray Water Seal.

B. Less than design bases spray flow because two 50% capacity pumps are operating providing spray flow and the Appendix J Containment Spray Water Seal.

C. Design bases spray flow is achieved because two 100% capacity pumps are operating providing full spray flow and the Appendix J Containment Spray Water Seal in the same loop.

D. Design bases spray flow is achieved because two 50% capacity pumps are operating providing spray flow and the Appendix J Containment Spray Water Seal.

K&A # 226001 A3.03
Importance Rating 2.8 (RO) / 2.8 (SRO)

QUESTION 64

K&A Statement: **A3.03** – Ability to monitor automatic operation of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: including: System Flow

Justification:

A. A is incorrect – The flooding of the northeast containment spray compartment will result in pumps 112 and 122 feeding both the primary and secondary containment spray loops. As described in Answer C this will meet the design bases requirements for containment spray flow and meets the Appendix J Water Seal requirements. This is a plausible distractor for those candidates that do not recognize the physical separation and valve normal valve alignment provides for adequate system performance with both pumps on one in operation.

B. B is incorrect – Each pump is rated at 100% of the required containment spray flow and would discharge into both spray headers and provide both the required spray flow and Appendix J Water Seal requirements. This is a plausible distractor for those candidates that do not recognize that each of the four Containment Spray pumps is a 100% capacity pump.

C. CORRECT C is correct –Each containment spray pump is rated for 100% flow. Pumps 112 and 122 are located in the northwest containment spray pump compartment and feed both the primary and secondary spray loops. Either of the two pumps provides full rated spray flow with the second pump required to meet the Appendix J Containment Spray Water Seal. The Containment Spray System is normally in standby, lined up for automatic start. Flow from one Containment Spray Pump (3600 GPM) (95% to the Drywell and the remainder to the Torus) is sufficient to remove the postulated post-accident core energy released.

D. D is incorrect – Each pump is rated at 100% of the required containment spray flow and discharge into both the primary and secondary spray loops which provide both the required spray flow and Appendix J Water Seal requirements. This is a plausible distractor for those candidates that do not recognize that each of the four Containment Spray pumps is a 100% capacity pump.

References: N1-OP-14, Section B Student Ref: None

Learning Objective: N/A

Question source: NMP1 2005 NRC Exam

Question History: NMP1 2005 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 65

With the plant at 100% power, a large crack in the outer wall of the steam dryer occurs as indicated on the drawing provided.

Per N1-SOP-1.5, Unplanned Reactor Power Change, which one of the following are the changes in the following parameter?

- Reactor Power (REACT PWR),
- Reactor Recirc Suction Temp (RR SUCT TEMP), and
- Moisture Content in the Steam Leaving the Reactor (CARRYOVER)

as a result of this failure?

	Reactor PWR	RR Suct. Temp	CARRYOVER
A.	Decrease (↓)	Increase (↑)	Decrease (↓)
B.	Increase (↑)	Increase (↑)	Increase (↑)
C.	Decrease (↓)	Decrease (↓)	Increase (↑)
D.	Increase (↑)	Decrease (↓)	Decrease (↓)

K&A # 290002 K3.07
Importance Rating 3.1 (RO) / 3.1 (SRO)

QUESTION 65

K&A Statement: **K3.07 -** Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: **Nuclear boiler instrumentation**

Justification:

A. CORRECT A is correct – Per N1-SOP-1.5, Flow Path A: the dryer crack represents an ABOVE TOP GUIDE crack with indications of reduced reactor power and increased Recirc suction temperature. Per N1-SOP-1.5, 5.3 the failure of the steam dryer at Quad Cities produced the following indications:

- Lowered Reactor Power due to lowering pressure inside the core shroud.
- Increase in moisture content in steam (carry over).

This failure will also cause additional heating of the annulus water raising the recirc pump suction temperature thereby lowering the Net Positive Suction Head (NPSH). Increased moisture carry over will occur due to steam bypassing the Dryer.

B. B is incorrect – This is a plausible distracter for candidates who believe the crack in the dryer will produce greater steam flow and therefore increased power in the reactor. This increased power could result in a drop in Recirc temperature due to increased feed flow.

C. C is incorrect - This is a plausible distracter for candidates who believe the crack in the dryer could result in a drop in Recirc temperature due to increased feed flow.

D. D is incorrect – This is a plausible distracter for candidates who believe the crack in the dryer will produce greater steam flow and therefore increased power in the reactor.

References: N1-SOP-1.5 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2005 NRC Exam

Question History: NMP1 2005 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.7

QUESTION 66

Given the following:

- The last watch you stood was day shift on May 1
- You have been on vacation and are preparing to assume the shift as the ATC RO on day shift on May 10

Per the conduct of operations requirements, which one of the following describes the Control Room Logs that MUST Be reviewed BEFORE assuming the shift?

A. All Control Room logs back to and including night shift on May 1. Anything less is not appropriate.

B. All Control Room Logs back to and including day shift on May 3. Further back is not required and anything less is not appropriate.

C. Just the day and night shift Control Room logs for May 9. Further back is not required and anything less is not appropriate.

D. All Control Room Logs back to and including day shift on May 7. Further back is not required and anything less is not appropriate.

K&A # G2.1.3
Importance Rating 3.7

QUESTION 66

K&A Statement: Knowledge of shift or short term relief turnover practices.

Justification:

A. Correct – Per GAP-OPS-01 , 3.4.5, Prior to assuming the shift, the oncoming SM, CRS, STA, CSO, and licensed Reactor Operators shall read applicable logs from the date shift duty was last performed.

B. Incorrect because the last 7 days is not the requirement, but plausible if the candidate does not know the requirement then the last 7 days seems reasonable.

C. Incorrect because the last 24 hours is not the requirement for reading control room logs, but plausible because there are other requirements related to the last 24 hours.

D. Incorrect because the last 72 hours is not the requirement, but plausible if the candidate does not know the requirement then the last 72 hours seems reasonable.

References: GAP OPS-01 3.4.5 Student Ref: None

Learning Objective: N/A

Question source: NMP1 NRC Retake 2005

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 67

During a refueling outage, recirculation loops are going to be isolated for maintenance. What is the limitation on isolating recirculation loops and what is the reason for that limitation.

A. Limitation - The suction and discharge valves of at least **TWO** recirculation loops shall be full open unless the reactor vessel is flooded to above the main steam line nozzles, or the steam dry and separator are removed.

   Reason - To ensure reactor water level instrument readings indicate actual level in the core region.

B. Limitation - The suction and discharge valves of at least **ONE** recirculation loop shall be full open unless the reactor vessel is flooded to above the main steam line nozzles, or the steam dry and separator are removed.

   Reason - To prevent thermal stratification while shutdown with fuel in the Reactor Vessel.

C. Limitation - The suction and discharge valves of at least **TWO** recirculation loops shall be full open unless the reactor vessel is flooded to above the main steam line nozzles, or the steam dry and separator are removed.

   Reason - To prevent thermal stratification while shutdown with fuel in the Reactor Vessel.

D. Limitation - The suction and discharge valves of at least **ONE** recirculation loops shall be full open unless the reactor vessel is flooded to above the main steam line nozzles, or the steam dry and separator are removed.

   Reason - To ensure reactor water level instrument readings indicate actual level in the core region.

K&A # 2.1.32
Importance Rating 3.8

QUESTION 67

K&A Statement: Ability to explain and apply system limits and precautions.

Justification:

A. Correct – Per N1-OP-1, at least two recirc loops must be full open unless the reactor vessel is flooded above the main steam lines or the steam dryer and separator are removed.

B. Incorrect because at least two recirculation loops must be full open. Plausible because having one recirc loop open would give adequate communication between the vessel and downcomer, however procedurally requires two.

C. Incorrect because the reason is incorrect for this precaution. The precaution that requires forced circulation through the core with recirc pumps or Vessel level above main steam lines with SDC in service and all recirc pump lines isolated is to prevent thermal stratification. Plausible because with no communication between the vessel and downcomer which would provide natural circulation, there would be thermal stratification without forced circulation of water through the core.

D. Incorrect because at least two recirculation loops must be full open. Plausible because having one recirc loop open would give adequate communication between the vessel and downcomer, however procedurally requires two.

References: N1-OP-1 D.1.0 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 68

- You are a licensed Reactor Operator. You have been on vacation for the previous 14 days. This is your first shift back on watch. Your schedule for the week is to work Monday, Tuesday, Wednesday night shifts 8pm-8am. Thursday and Friday are days off. You are scheduled to work day shift 8am-8pm Saturday, Sunday and Monday.

- The plant is conducting a Reactor Startup after a plant trip on the previous Friday.
- You stay a total of 2 hours past your normal time on Monday due to an extensive turnover.

Which of the following will satisfy the requirements of GAP-FFD-02, Control Of Working Hours for you to work your scheduled shifts?

A. Notify your supervisor that your work hour limits will be exceeded and have the overtime deviation request approved by the Plant General Manager prior to working your shift on the following Monday.

B. No action required, because turnover time is not included in calculating working hours.

C. Notify your supervisor that your work hour limits will be exceeded and obtain verbal approval from your supervisor to work your scheduled hours.

D. Notify your supervisor that your work hour limits will be exceeded and have the overtime deviation request approved by the Plant General Manager prior to working your shift on Tuesday.

K&A # 2.1.5
Importance Rating 2.9

QUESTION 68

K&A Statement: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Justification:

A. Incorrect because the request has to be approved in advance of exceeding limits. The first limit exceeded is working greater than 24 hours in any 48 hour period, which is exceeded on Tuesday.

B. Incorrect because turnover time of greater than 1 hour is counted towards calculating work hours.

C. Incorrect but plausible because this is the correct response for personnel who do NOT perform or directly supervise those who perform safety related functions.

D. Correct - because the request has to be approved in advance of exceeding limits. The first limit exceeded is working greater than 24 hours in any 48 hour period, which is exceeded on Tuesday.

References: GAP-FFD-02 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 41.10

QUESTION 69

Which one of the following requirements must be met prior to performing manipulation of tagged components under an Operating Permit Tag?

A. The Tagging Authority is required to obtain concurrence from all Work Order Holders.

B. The Operating Permit Section Holder is required to obtain concurrence from all Work Order and Worker Tracking List Holders.

C. The Tagging Authority is required to obtain concurrence from all Operating Permit Section Holders and Work Order Holders.

D. The Operating Permit Section Holder is required to obtain concurrence from all Worker Tracking List Holders.

K&A # 2.2.13
Importance Rating 4.1

QUESTION 69
K&A Statement: Knowledge of tagging and clearance procedures.
Justification:

A. Incorrect because the tagging authority gives authorization to the section holder to manipulate components, but the section holder needs concurrence from the work order and worker tracking list holders.

B. Correct – 5.4.H of CNG-OP-1.01-1007 states the Operating Permit Section Holder is required to obtain concurrence from all Work Order and Worker Tracking List Holders prior to performing manipulation of tagged components.

C. Incorrect because the tagging authority gives authorization to the section holder to manipulate components, but the section holder needs concurrence from the work order and worker tracking list holders.

D. Incorrect because the Section Holder needs both the work order and worker tracking list holders.

References: CNG-OP-1.01-1007 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 70

WHICH ONE of the following describes a condition that will violate a Unit 1 Technical Specification Safety Limit?

A. Drywell pressure rises to 70 psig

B. Reactor level drops to -20 inches

C. Reactor pressure rises to 1370 psig

D. Minimum Critical Power Ratio (MCPR) lowers to 1.10

K&A # 2.2.22
Importance Rating 4.0

QUESTION 70

K&A Statement: 2.2.22– Knowledge of limiting conditions for operations and safety limits

Justification:

A. Incorrect- Drywell pressure is not a safety limit but plausible since it exceeds the maximum design pressure for the drywell.

B. Correct – Although the limit is applicable when shutdown, the plant will scram at on decreasing level and be shutdown. -10 inches level indicated is the safety limit.

C. Incorrect – Reactor pressure rising to 1370 psig does not exceed a Safety Limit but plausible since it is above the Safety valve setpoint.

D. Incorrect Plausible since MCPR was not less than the Safety Limit Critical Power Ratios of 1.07.

References: Tech Spec. 2.2.1, 2.2.1, COLR Student Ref: None

Learning Objective: N/A

Question source: Limerick 08 NRC exam

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.5

QUESTION 71

The plant is at rated power when the following annunciator alarms:

- H1-1-7 OFF GAS HIGH RADIATION
- The reactor operator reports that both channels of the Offgas Radiation Monitors are at the alarm setpoint.

Which one of the following states the required actions per N1-SOP-25.2, Fuel Failure or High Activity in the Rx Coolant or Off Gas?

A. Confirm Isolation of the Off Gas System

B. Scram the Reactor per N1-SOP-1, Reactor Scram

C. Reduce Reactor Power as necessary to control radiation levels

D. Verify mechanical pump trip and isolation

K&A # 2.3.11
Importance Rating 3.8

QUESTION 71
K&A Statement: Ability to Control Radiation Releases
Justification:

A. Incorrect because the isolation has not occurred when the high offgas alarm comes in, but plausible if the candidate thinks the system has isolated.

B. Incorrect because sop-25.2 only directs a scram if main steam line monitors are 3.75 times normal full power background, but plausible if the candidate does not realize this action is only for high main steam line rad monitors.

C. Correct – With an off gas high rad alarm in, sop-25.2 directs to reduce power to control radiation levels.

D. Incorrect because the step to verify mechanical pump trip is off the main steam high rad leg of the sop-25.2, but plausible if the candidate does not realize this action is only for high main steam line rad monitors.

References: N1-SOP-25.2 Student Ref: None

Learning Objective: N/A

Question source: OC 2007 NRC

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.11

QUESTION 72

The Main Control Room has been abandoned due to a fire:

- Operators are required to enter a posted radiation area in order to manually operate Primary Containment Isolation Valves.
- The highest dose in the area is 1500 mR/hr.

This area is required to be posted as a ____(1)____ and personnel entering the area are required to ___(2)____.

	(1)	(2)
A.	High Radiation Area	maintain each entrance closed and locked except when inside the high radiation area so personnel are not prevented from leaving the area.
B.	Locked High Radiation Area	maintain each entrance closed and locked except when inside the high radiation area so personnel are not prevented from leaving the area.
C.	Locked High Radiation Area	maintain each entrance closed and locked except for periods of ingress or egress, unless guarded to prevent unauthorized entry.
D.	Very High Radiation Area	maintain each entrance closed and locked except for periods of ingress or egress, unless guarded to prevent unauthorized entry.

K&A # G2.3.13
Importance Rating 3.4 (RO) / 3.8 (SRO)

QUESTION 72

K&A Statement: **G2.3.13** – Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Justification:

A. A is incorrect – The area radiation reading is greater than the threshold for a high radiation area but is also above the locked high radiation area limit so posting at the lower level would be incorrect. Although barriers must be designed to ensure personnel are not prevented from leaving the area (GAP-RPP-08 Step 3.1.4); this does not mean that they may be left unlocked to achieve this goal. This is a plausible distractor for those candidates that misinterpret the requirement to not restrict egress from the area as the ability to maintain the opening unlocked.

B. B is incorrect – The area radiation reading >1000 mR/hr require posting this area as a locked high radiation area. The entrance to each locked high radiation area must remain closed and locked as described below in Answer C. Although barriers must be designed to ensure personnel are not prevented from leaving the area (GAP-RPP-08 Step 3.1.4); this does not mean that they are to be left unlocked to achieve this goal. This is a plausible distractor for those candidates that misinterpret the requirement to not restrict egress from the area as the ability to maintain the opening unlocked.

C. CORRECT C is correct – The area radiation reading >1000 mR/hr require posting this area as a locked high radiation area. The entrance to each locked high radiation area must remain closed and locked except periods of ingress and egress unless the entrance is guarded to prevent unauthorized entry (GAP-RPP-08, Step 3.1.2 and 3.6.1.c).

D. D is incorrect – The area radiation readings do not require posting as a Very High Radiation Area (500 R/hr at one meter). The controls in Item 2 are correct in this answer. This is a plausible answer for a candidate that incorrectly classifies the radiation area classification required at the specified dose rate.

References: GAP-RPP-08 Student Ref: None

Learning Objective: N/A

Question source: NEW

Question History: NEW

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.12

QUESTION 73

A plant transient has occurred on Unit 1 with the following conditions now present:

- Reactor Level – 63 inches and slowly rising
- Reactor Pressure – 1090 psig and steady
- Reactor Power – 50% and steady
- Drywell Pressure – 3 psig and slowly rising
- Drywell Temperature – 145F and slowly rising
- Torus Level – 11.25 feet and slowly rising
- Torus Temperature – 82F and slowly rising
- H2-2-2, R BUILDING EQUIP DRAIN LEVEL HIGH, is in alarm

What are the appropriate procedures to be entered based on the given plant conditions?

A. EOP-2, EOP-3, EOP-4, EOP-5

B. EOP-2, EOP-4

C. EOP-4

D. EOP-2, EOP-3, EOP-4

K&A # G2.4.1
Importance Rating 4.6 (RO) / 4.8 (SRO)

QUESTION 73

K&A Statement: **G2.4.1** – Knowledge of EOP entry condition and immediate action steps

Justification:

A. A is incorrect – The scram condition exists with reactor pressure above the reactor scram setpoint of 1080 psig meets the entry condition for EOP-2; however the failure to scram demonstrated by the current reactor power requires entry into EOP-3 as well. EOP-4 is entered due to current Torus level of 11.25 feet. The conditions for EOP-5 are not met with the equipment drain tank level high alarm. This is a plausible distractor for those candidates that do not discern between the floor and equipment side drain tank levels for EOP-5 entry.

B. B is incorrect – A scram condition exists with reactor pressure above the reactor scram setpoint of 1080 psig and EOP-2 must be entered; however the failure to scram demonstrated by the current reactor power level leads to an entry into EOP-3 as well. EOP-4 is entered due to current Torus level of 11.25 feet. This is a plausible distractor for those candidates that do not recognize ATWS that is in progress and EOP-3 entry condition.

C. C is incorrect – The conditions for EOP-5 are not met with the equipment drain tank level high alarm. The entry condition to EOP-5 is based on floor drain tank level high. This is a plausible distractor for those candidates that do not recognize the scram condition on 1090 psig reactor pressure. Missing the scram condition will lead to the candidate not identifying the EOP-2 or EOP-3 entry conditions.

D. CORRECT D is correct – Two entry conditions for EOP-2 exist with (1) reactor pressure above the scram setpoint of 1080 psig and (2) reactor power at 50% after the scram. However the failure to scram also requires entry into EOP-3. EOP-4 is executed due to the current Torus level of 11.25 feet which meets the entry criteria.

References: EOP-2, EOP-3, EOP-4, EOP-5 Student Ref: None

Learning Objective: N/A

Question source: NEW

Question History: NEW

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.10

QUESTION 74

The plant is operating at power, with the following:

- Shift Manager has declared an UNUSUAL EVENT due to Seismic Event
- The extra Reactor Operator normally assigned to the WEC is dispatched to the Reactor Building to perform damage control actions
- While in the Reactor Building, a station announcement is made upgrading the event to an ALERT

Which one of the following actions is required by the extra RO, per EPIP-EPP-22, Damage Control?

A. Immediately report to the WEC and wait to be assigned to a repair team.

B. Immediately report to the OSC and wait to be assigned to a repair team.

C. Contact the Shift Manager from the Reactor Building for directions.

D. Contact the OSC Coordinator from the Reactor Building for directions.

K&A # G2.4.29
Importance Rating 3.1

QUESTION 74
K&A Statement: Knowledge of the emergency plan.
Justification:

A. Incorrect but plausible because the WEC would be one area to assemble and wait for directions until the OSC is manned.

B. Incorrect but plausible because eventually the DC teams will be dispatched out of the OSC.

C. Correct – The procedure directs personnel dispatched as DC to contact the SM for directions when the announcement of staffing of the OSC is made. The OSC will be manned when going to the Alert.

D. Incorrect but plausible because eventually the DC teams will report to the OSC coordinator.

References:	EPIP-EPP-22 3.9	Student Ref:	None
Learning Objective:	N/A		
Question source:	NMP1 2005 NRC Retake Exam		
Question History:	None		
Cognitive level:	Memory/Fundamental knowledge: Comprehensive/Analysis:	X	
10CFR	41.10		

QUESTION 75

The plant is operating at 100% power. The following alarm window annunciates:

- F2-1-2, REACT RECIRC PUMP-MOTOR 12

The 12 Recirc Pump seal indications are as follows:

- Seal #1 pressure 380 psig
- Seal #2 pressure 350 psig

Which one of the following resulted in the above alarm and indications on the 12 Recirculation Pump?

A. Only Seal #1 has failed

B. Breakdown orifice between the seals has plugged.

C. Only Seal #2 has failed.

D. Both Seal #1 and Seal #2 have failed

K&A # G2.4.31
Importance Rating 4.2

QUESTION 75

K&A Statement: Knowledge of annunciator alarms, indications, or response procedures.

Justification:

A. Incorrect but plausible since the pressures would be about equal in both seal #1 and seal #2 if seal #1 had failed. However the pressure would have been much higher (approx 1000psig)if seal #1 had failed.

B. Incorrect but plausible since the pressure would remain high on #1 seal however #2 seal would have a lower pressure if the breakdown orifice is plugged.

C. Incorrect but plausible since the pressure would have remained 1000 psig on #1 seal and #2 seal would have been lower than 500 psig if only #2 seal failed. if the

D. Correct – Normal Pressure on the #1 and #2 seals would be 1,000 psig and 500 psig respectively. Approximately equally low pressure on both seals is an indication of failure of both pump seals.

References:	N1-SOP-1.2, N1-OP-1	Student Ref:	None

Learning Objective: N/A

Question source: Lim 08 NRC

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 76

The unit was shut down due to rising drywell pressure.

One (1) minute after the scram, the following conditions exist:

- RPV water level indicates 25 inches on 36-09 and 36-10
- RPV water level indicates 38 inches on 36-76A and 36-77A
- RPV water level indicates 0.5 feet on the Wide Range
- RPV water level indicates 4 inches on both Lo-Lo-Lo Rosemounts
- RPV water level indicates 10 inches on both Fuel Zone Indicators
- Reactor pressure is 425 psig and lowering quickly
- Drywell pressure is 13 psig and rising quickly
- Drywell temperature is 220°F and rising quickly
- Both Core Spray Loops are in service providing make-up
- Recirc Pumps 14 and 15 were shutdown in preparation for shutdown cooling operations. All other pumps remain in service.

Based on the above, which one of the following is required?

A. Enter EOP-7, RPV Flooding

B. Enter EOP-8, RPV Blowdown

C. Initiate Containment Sprays

D. Restore and Maintain Level between 53" and 95"

K&A #	295031 A2.01
Importance Rating	3.8 (RO) / 4.1 (SRO)

QUESTION 76

K&A Statement: A2.01 – Ability to determine and/or interpret the following responses as they apply to REACTOR LOW WATER LEVEL: Reactor Water Level.

Justification:

A. CORRECT A is correct – Current plant conditions have left all EOP level indication unavailable for use by the control room. The indications are unavailable for the following reasons per Figure A of EOP-2:

GEMAC and Yarway level indication (36-09, 36-10, 36-76A and 36-77A) are showing downscale (less than zero). The current drywell temperature allows use of the HI/LO or LO-LO level instrumentation when it is indicating greater than 0 inches. However, the rapid depressurization of the vessel to 425 psig makes these instruments unreliable and should not be used for level indication.

Wide range level indication (36-94) is not usable when indicating less than 1.5 feet at the current drywell temperature. With the current 0.5 foot reading this indicator may not be used.

Lo/Lo/Lo Level indications (36-19, 36-20) are not usable with both loops of Core Spray in operation.

Fuel Zone indication (36-43, 36-44) is not usable with Recirc Pumps still in operation following the shutdown.

B. B is incorrect – Reactor level is unknown at this time. Given the conditions above all of the available instrumentation in unavailable and EOP-7 Reactor Flooding is required to be entered. EOP-8, Emergency blowdown is not required to be entered unless the conditions of EOP-7 are not able to be met. Since the actions required by EOP-7 have not yet been attempted an emergency blowdown is not warranted.

C. C is incorrect – Current plant conditions do not meet the criteria to initiate containment sprays. This would have a positive effect on containment temperature but is not allowable under the current set of drywell pressure and temperature conditions.

D. D is incorrect – With Reactor level unknown the actions of EOP-7 must be met which overrides the direction of EOP-2 which restores level to a normal band of 53 to 95 inches.

References: N1-EOP-2, N1-EOP-7 Student Ref: Fig. A

Learning Objective: N/A

Question source: Modified NMP Bank

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.10

QUESTION 77

A steam leak in the drywell has occurred.

Primary containment parameters for the past four minutes are as follows:

	08:01	08:02	08:03	08:04
Drywell Pressure	6.0 psig	8.0 psig	11.0	14.0
Torus Pressure	5.0 psig	7.0 psig	10.0	13.0
Drywell Temperature	225 F	250 F	276 F	302 F

Which one of the following is the EARLIEST TIME at which containment spray can be initiated?

A. 08:01

B. 08:02

C. 08:03

D. 08:04

K&A # 295028 A2.04
Importance Rating 4.2

QUESTION 77

K&A Statement: Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:
Drywell pressure.

Justification: K/A matches because question requires interpreting the relationship between high drywell temperature and the high drywell pressure and determining the mitigation strategy required for these conditions.

A. Incorrect because at 225 F, drywell pressure must be above ***6.8*** psig to spray but plausible if the candidate does not read the curve correctly.

B. Correct - At 250 F, drywell pressure must be above 7.3 psig to spray. This requirement is met for these conditions. All of the other conditions are incorrect.

C. Incorrect because at 276 F, drywell pressure must be above 7.8 psig to spray. This requirement is met for drywell pressure but plausible because if the candidate did not pick B then this would be the next time that could spray but not the earliest.

D. Incorrect because at 302 F, drywell pressure must be above 8.4 psig to spray. This requirement is met for these conditions however the "okay to spray" region was entered earlier. Plausible if the candidate thinks must wait until hit 13 psig.

References: EOP-4 Student Ref: EOP-4

Learning Objective: N/A

Question source: NMP1 2004 NRC

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.5

QUESTION 78

The plant is in a refueling outage with Fuel Shuffle Part II in progress.

Shutdown Cooling (SDC) is lost.

Which one of the following is required if actual temperature measurements indicate reactor water temperature is projected to exceed 140°F?

A. Perform a time to boil estimation.

B. Verify Secondary Containment established.

C. Perform a feed and bleed to/from the Fuel Pool

D. Return to nearest storage location in Spent Fuel Pool <u>OR</u> Core, core components being transferred.

K&A # 295021 AA2.04
Importance Rating 3.6

QUESTION 78

K&A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water temperature.

Justification:

A. Correct – Per N1-SOP-6.1 if the safety assessment level for DHR <N+1 and temperature is predicted to exceed 140F, then perform a time to boil.

B. Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1.

C. Incorrect but plausible because N1-SOP-6.1 has a step to perform feed/bleed on the SFP if there is no fuel in the vessel. The stem states that fuel movements have not started, therefore there is fuel in the vessel.

D. Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1.

References: N1-SOP-6.1 Student Ref: None

Learning Objective: N/A

Question source: NMP1

Question History: NRC 2004 Exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.5

QUESTION 79

A loss of coolant accident has occurred, with plant conditions as follows:

- * Containment Spray Pump 111 is clearance tagged out of service
- * PB 103 is deenergized due to a fault
- * All available systems are injecting
- * RPV water level is -70 inches and steady
- * RPV Pressure is 200 psig and lowering rapidly
- * Torus pressure is 18 psig and slowly lowering
- * Torus temperature is 130 F and slowly rising
- * Torus level is 12 feet and stable
- * Drywell Pressure is 21 psig and slowly lowering
- * Drywell Temperature is 250 F and slowly lowering
- * Containment Spray Pump 112 has just tripped on over-current

Which one of the following is required?

A. Vent the Torus per EOP-4.1, Section 2

B. Cool the Torus per EOP-1, Attachment 16

C. Operate Containment Sprays using raw water through Loop 11 sparger per EOP-1, Attachment 17

D. Blowdown per EOP-8 while continuing in EOP-4

K&A #	295024 G2.1.23
Importance Rating	4.0

QUESTION 79

K&A Statement: 2.1.23. Ability to perform specific system and integrated plant procedures during different modes of plant operation, as it relates to HIGH DRYWELL PRESSURE.

Justification:

A. A is incorrect - Conditions to vent PC do not exist for the current Pressure Suppression Pressure conditions. Venting of primary containment is on EOP-4 but comes after attempts to spray the drywell.

B. B is incorrect - Requires a Containment Spray pump available to operate in torus cooling. The loss of PB 103 has eliminated the use of Containment Spray Pumps 121 and 122. CS Pump 111 is currently tagged out of service and CS pump 112 tripped off. This condition eliminates all available pumps.

C. CORRECT - Per EOP-4, Step PCP-4 can use external source (raw water) to spray as long as you stay below PCPL.

D. D is incorrect – The conditions for blowdown do not exist in the Primary Containment Pressure, Drywell Temperature, and Torus Water Level legs of EOP-4. Additionally, reactor level is above the blowdown required limit from EOP-2 so the conditions for blowdown on Reactor level are also not met. Blowdown of the reactor on Pressure Suppression pressure comes after the step to spray the drywell using external water sources in EOP-4.

References: N1-EOP-4 Student Ref: EOP-4

Learning Objective: N/A

Question source: Modified NMP1 2002 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.10

QUESTION 80

RPV Depressurization **IS REQUIRED**

Which set of conditions below would potentially result in Primary Containment failure.

A. RPV Pressure of 950 psig
Torus Temperature of 120 F
Torus Water Level of 11.5 feet

B. RPV Pressure of 700 psig
Torus Temperature of 130 F
Torus Water Level of 11 feet

C. Drywell temperature of 350 F
Drywell pressure of 3 psig

D. Torus pressure of 30 psig
Primary Containment Water Level of 360 inches

K&A # 295025 G2.4.21
Importance Rating 3.9

QUESTION 80

K&A Statement: **G2.4.21-Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.,** as it relates to High Reactor Pressure.

Justification: Justification for K/A match. Several parameters are used to assess the status of containment conditions for this question. Although Reactor Pressure is not above normal operating pressure, the Reactor Pressure is high for the given torus temperature and level and therefore the K/A matches.

A. CORRECT A is correct – the current conditions in Answer A show that the HCTL is violated and further addition of heat to the Torus places the containment in a condition where initiation of RPV depressurization via ERV's will result in the primary containment pressure limit being reached before the rate of energy transfer from the RPV is within the capability of either a single 12" (or larger) containment vent or the hardened vent. Therefore initiation of ERV's for depressurization will progress further towards primary containment failure not prevent containment failure.

B. B is incorrect – HCTL is not violated in answer B. In this condition depressurization with ERV's is the desired protective strategy. Additional depressurization with ERV's will allow HCTL to remain below the HCTL limit for the current Torus level and aids in the prevention of a containment failure. Answer B is incorrect.

C. C is incorrect – Conditions are on the BAD side of the Containment Spray Initiation Curve, which prevents initiation of containment sprays. Depressurization via normal cooldown or blowdown if the provided conditions are not changed will protect the containment by removing energy from the reactor and preventing over pressurization of the containment in the event of a LOCA. Answer C is incorrect.

D. D is incorrect – The conditions in Answer D place the plant on the good side of the Primary Containment Pressure Limit (PCPL). PCPL is based only on the structural considerations of containment. RPV depressurization has no impact on the acceptability of containment conditions when evaluated against PCPL. Answer D is incorrect.

References: N1-EOP-4 Student Ref: EOP-4

Learning Objective: N/A

Question source: Modified OC 2007 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5

QUESTION 81

The plant is operating at 100% power when a significant fire occurs in the Main Control Room and procedure N1-SOP-21.2 is entered.

The following events occur as a results of the fire:

- The only action taken prior to evacuating the Main Control Room was placing the Reactor Mode Switch in SHUTDOWN
- A wiring error allowed the fire to cause ERV's 111 and 112 to spuriously open until their fuses were pulled per N1-SOP-21.2.
- Reactor level dropped due to the ERV actuation until an automatic Vessel and Containment Isolation occurred.
- Reactor level was recovered using Core Spray.
- All required actions of N1-SOP-21.2 are being implemented by the SM, CRS, ATC and In-Plant RO.

Based on the above conditions, what is the requirement for notifying the NRC of this event?

A. 15 minutes

B. 1 hour

C. 4 hours

D. 8 hours

K&A # 295016 G2.4.30
Importance Rating 4.1 (SRO)

QUESTION 81

K&A Statement: **G2.4.30**-Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, | the NRC, or the transmission system operator, as it relates to **Control Room Abandonment**

A. A is incorrect – There are no 15 minute reporting requirements within CNG-NL-1.01-1004. This is a plausible distractor for those candidates that recognize the Emergency classification condition and then select the reporting requirement per the Emergency Plan and not the Regulatory Reporting requirement asked for in the stem.

B. CORRECT B is correct – The evacuation of the Control Room requires a declaration of an ALERT in the Emergency Plan under Item 7.2.2. The declaration of any of the Emergency Classes is reportable within 1 hour under 50.72(a)(1)(i) as described on page 3 of CNG-NL-1.01-1004. This is the most limiting report required to be made for this event. The injection of Core Spray into the reactor to control level and manual scram and RPS actuations both require a 4 hour ENS report under 50.72(b)(2)(iv)(A) and 50.72(b)(2)(iv)(A) (Section 9, page 7). The Automatic isolations that occurred on +5" reactor level require an 8 hour report under 50.72(b)(3)(iv) (Section 9, Page 8). The wiring error that caused level and pressure to drop may be reported under 50.72(b)(3) as an event that prevented the fulfillment of the system from performing it's safety function. This is also an 8 hour report. There are no 15 minute reporting requirements within CNG-NL-1.01-1004.

C. C is incorrect – The injection of Core Spray into the reactor to control level and manual scram and RPS actuations both require a 4 hour ENS report under 50.72(b)(2)(iv)(A) and 50.72(b)(2)(iv)(A) (Section 9, page 7). This is not the most limiting report required to be made. This is a plausible distractor for those candidates that do not recognize the Emergency Classification that must be made with this event.

D. D is incorrect – The Automatic isolations that occurred on +5" reactor level require an 8 hour report under 50.72(b)(3)(iv) (Section 9, Page 8). The wiring error that caused level and pressure to drop may be reported under 50.72(b)(3) as an event that prevented the fulfillment of the system from performing it's safety function. This is also an 8 hour report. This is not the most limiting report required to be made. This is a plausible distractor for those candidates that incorrectly assess the repoting requirements for this event.

References: CNG-NL-1.01-1004, N1-SOP-21.1, N1-SOP-21.1 Student Ref: CNG-NL-1.01-1004

Learning Objective: N/A

Question source: Fitz 2007 NRC Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 Comments: 41.10

QUESTION 82

The plant is operating at 100% power when the Fire Detection System senses a fire in Hazard C-2151, D/G 103 on Elevation 261'.

The following Alarm Detection Zones are received at the Main Fire Control Panel:

- DX-2151A
- DX-2151B

An operator in the area reports:

- The Local horn and light have actuated
- C02 has discharged into the area

In response to the above conditions, which one of the following is required at this time?

A. Ensure Control Room is at a positive pressure relative to outside air pressure or initiate pressurization mode of control room ventilation.

B. Dispatch a single fire brigade member to the alarm location to verify the fire condition before dispatching the fire brigade.

C. Direct the CSO to implement EPIP-EPP-28, Attachment I, CSO Fire Fighting Checklist.

D. Declare an unusual event per EAL 8.2.1 and enter EPIP-EPP-18, Activation and Direction of the Emergency Plans.

K&A #	600000 G2.1.2
Importance Rating	4.4 (SRO)

QUESTION 82

K&A Statement: **G2.1.2** - Knowledge of operator responsibilities during all modes of plant operation: **Plant Fire on Site**

A. A is incorrect – This is the required action when CO2 or Halon discharges in rooms which are adjacent to the main control room. Since the EDG 261' elevation is not contiguous to the control room this action is not required. This is a plausible distractor for those candidates that do not recognize this action is only required for those areas which are contiguous with the control room.

B. B is incorrect – This is the correct action if alarms are not associated with the actuation of an automatic suppression system within the protected area. Because automatic suppression occurred, the fire brigade (not just a single member) is dispatched to the scene. This is a plausible distractor for those candidates that do not recognize the change is required response with the automatic actuation of the CO2 system.

C. CORRECT C is correct – When credible evidence exists of a fire condition within the protected area, then per EPIP-EPP-28, direct the CSO to implement the CSO fire fighting checklist. The definition of CONFIRMED FIRE is a condition in which credible evidence exists that a fire is actually occurring. A fire may be considered as confirmed given any of the following: fire alarm/annunciator AND suppression system activation accompanied by actual flow or discharge, or Fire Brigade/Leader report, or SSS judgement.

D. D is incorrect – An unusual event is not declared until confirmed fire not extinguished within **15** minutes of control room notification. There is no evidence that the fire is not extinguished and 15 minutes have not expired. This is a plausible distractor for those candidates who believe that the declaration must be made at the time of verification of the fire.

References:	EPIP-EPP-28, N1-OP-21C, N1-SOP-21.1	Student Ref:	None

Learning Objective:	N/A	
Question source:	Modified NMP1 2004 Exam	
Question History:	NMP1 2004 Exam	
Cognitive level:	Memory/Fundamental knowledge:	X
	Comprehensive/Analysis:	
10CFR 55	41.10	
Comments:		

QUESTION 83

The plant is at 80% power when a seismic event occurs. Subsequently, because of a torus water leak the following are observed:

- H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, in alarm
Computer Pt. F188 NE RB CORNER RM WTR LVL HIGH in alarm

Which one of the following is the operability of the safety-related pumps in this area?

A. Core Spray Pumps 121 and 122 are inoperable at this time.

B. Containment Spray Pumps 112 and 122 are inoperable at this time.

C. Core Spray Pumps 121 and 122 remain operable until level in this area rises an additional 2 feet.

D. Containment Spray Pumps 112 and 122 remain operable until level in this area rises an additional 2 feet.

K&A # 295036 A2.01
Importance Rating 3.2

QUESTION 83

K&A Statement: **Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : Operability of components within the affected area..**

Justification:

A. Incorrect because the Core Spray Pumps 121 and 122 are in the SE corner room and are not affected. But plausible if the candidate does not know which components are in this room.

B. Correct - Containment Spray Pumps 112 and 122 are in the NE corner room and are the components affected by the water level in the room. The alarm is actuated at a water level of 5 feet in the room, which is the maximum safe value. The max safe value is defined to be the highest value at which equipment necessary for the safe shutdown of the plant will operate. Therefore the components in the area are inoperable.

C. Incorrect the maximum safe value is already reached. Core Spray Pumps 121 and 122 are in the SE corner room and are not affected But plausible if the candidate does not know which components are in this room.

D. Incorrect The maximum safe value is already reached. The alarm is actuated at a water level of 5 feet in the room, which is the maximum safe value. The max safe value is defined to be the highest value at which equipment necessary for the safe shutdown of the plant will operate. Therefore the components in the area are inoperable.

References: ARP H2-2-1 Student Ref: None

Learning Objective: N/A

Question source: NMP1

Question History: NRC 2004 Exam

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.5

QUESTION 84

The plant is at 100% power with fuel leaks identified:

1500 K1-1-2, EMER COND VENT 11 RAD MONITOR, alarms
1501 EMERG COND RMON 111 and 112 on J panel are both in alarm
1502 CRS directs EC 11 be isolated but it CANNOT be isolated
1505 EMERG COND RMON 111 and 112 are steady at 28 mremlhr
1506 Manual reactor scram inserted
1507 EMERG COND RMON 111 and 112 are at 15 mremlhr and lowering slowly

Which one of the following is the required action AT THIS TIME?

A. Cooldown at a rate below 100 F/hr as directed by EOP-2.

B. Cooldown at a rate above 100 F/hr as directed by EOP-2.

C. Perform a RPV Blowdown per EOP-8 as directed by EOP-5.

D. Perform a RPV Blowdown per EOP-8 as directed by EOP-6.

K&A # 295017
Importance Rating 4.2

QUESTION 84

K&A Statement: **A2.01**-Ability to determine and/or interpret the following as they apply to High Off-site Release Rate: **Off-site Release Rate: Plant specific**.

Justification:

A. Correct – No conditions at this time would warrant exceeding a normal cooldown.

B. Incorrect because conditions do not warrant exceeding tech spec cooldown rates. Plausible if candidate is anticipating blowdown, however with lowering rad levels this would not be the correct action.

C. Incorrect because although EOP-5 may be entered, a primary system is not discharging into the Reactor Building and therefore a blowdown is not the correct action. Plausible if candidate does not make the distinction that a primary system is not discharging into the rx building.

D. Incorrect because EOP-6 is entered on an alert and the rad monitors do not make it to the alert level. Plausible if the candidate thinks that he is in eop-6 but for blowdown prior to GE levels he would need more info to determine if blowdown is required.

References: EOP 2, 5 & 6 Student Ref: EOP 2, 5 & 6

Learning Objective: N/A

Question source: NMP1 2004 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.5

QUESTION 85

There is a leak into secondary containment. EOP-5 Secondary Containment Control has been entered. Evaluate each set of conditions and leak source below.

EOP-5 step SC-5 states, Isolate all discharges into affected areas except systems needed for:
Damage Control
Other EOP actions.

In which one of the following situations, should the leaking system or component be left in service?

A. Emergency Condenser 11 has a steam leak on isolation valve 39-05. The MSIV's inadvertently isolated during the transient. CRD has tripped and is not able to be put back in service. EC 12 is blocked out for maintenance.

B. RWCU is in recirculation mode per EOP-1, Attachment 9 with a steam and water leak on suction isolation valve 33-04. An ATWS is in progress with the MSIV's closed and torus level at 6 feet and lowering. Both EC's are in service with reactor pressure 950 psig and steady.

C. Emergency condenser 12 has a steam and water leak from a shell side fault due to a tube rupture. Reactor level is 75 inches and dropping slowly. The MSIV's inadvertently isolated during the transient. No other sources of high pressure feed are available.

D. RWCU is in reject mode per EOP-1, Attachment 8 with a steam and water leak on FCV-33-160. An ATWS is in progress with the MSIV's closed and torus level at 11 feet and steady. Both EC's are in service and reactor pressure is 850 psig and slowly rising.

K&A # 295032 G2.4.6
Importance Rating 4.7 (SRO)

QUESTION 85

K&A Statement: G2.4.6 –Knowledge of EOP mitigation strategies as related to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE.

Justification:

A. A is incorrect – With the MSIV's isolated and EC 12 out of service pressure control can be transitioned to ERV's and EC 11 can be removed as a pressure control system by isolating EC 11. Level control can be maintained with HPCI. Plausible if the candidate does not realize that HPCI is available.

B. B is correct – With RWCU in recirculation mode per EOP-1, RWCU is being used as pressure control system. Both EC's are in service and pressure is steady under this condition. No other pressure control mechanisms are available for these conditions with the MSIV's closed and torus level being too low to utilize ERV's. Therefore RWCU must remain in service to stabilize reactor pressure and can not be taken out of service.

C. C is incorrect. With the reactor isolated, the condenser unavailable for pressure control and no high pressure feed available for level control only EC's are available to control level. Pressure may be transitioned to EC11 or ERV's under the current condition. The leak on the EC Loop 12 can be isolated and EC Loop 11 placed in service to maintain plant conditions. Plausible if candidate does not realize has EC 11 available.

D. D is incorrect – ERV's are available for pressure control and therefore RWCU can be removed from service and isolated. Plausible because with pressure rising the candidate may think RWCU should not be removed. But with alternate pressure control mechanisms available the leak should be isolated.

References: EOP-4, 5, 3 Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 43.5

QUESTION 86

Technical specifications require a minimum amount of fuel supply onsite for EDG operation.

Which one of the following are the bases for the minimum fuel supply for EDG Operation.

A. Both EDG's providing the required power to equipment following a LOCA for two days.

B. A single EDG providing the required power to equipment following a LOCA for two days.

C. Both EDG's providing the required power to equipment following a LOCA for four days.

D. A single EDG providing the required power to equipment following a LOCA for four days.

K&A # 262001 G2.2.25
Importance Rating 4.2

QUESTION 86

K&A Statement: **Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.**

Justification:

A. Incorrect because the minimum fuel required is for one not 2 EDG's but plausible if the candidate does not know the basis.

B. Correct – The tech spec is that a minimum 2 day supply for one EDG is onsite. 11,650 gallons is the amount that one EDG will need to supply the required power to equipment following a LOCA.

C. Incorrect because the minimum fuel required is for one not 2 EDG's but plausible if the candidate does not know the basis.

D. Incorrect because the bases is 2 days not 4 days. Plausible because both EDG fuel oil tanks combined would give 4 days supply.

References: TS 3.6.3 Bases Student Ref: None

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.2

QUESTION 87

The plant is operating, with the following:

- Differential pressure transmitter 36-06A has failed its surveillance calibration check
- The failure is non-conservative

Which one of the following describes the action required by Technical Specifications?

A. Place the inoperable channel in the tripped condition within 24 hours.

B. Commence a normal shutdown within one hour, and be in cold shutdown within 10 hours.

C. Place the inoperable channel in one trip system in the tripped condition within one hour, and place the inoperable channel in the other trip system in the tripped condition within 24 hours.

D. Restore operability of the affected system within 7 days, and perform the required additional surveillances on the remaining operable system until then.

K&A # 207000 G2.2.40
Importance Rating 4.7

QUESTION 87

K&A Statement: Ability to apply Technical Specifications for a system as it relates to isolation condensers.

Justification:

A. Incorrect but plausible if the candidate does not read note a which explains that the d/p switches provide inputs to one channel in each trip system and therefore thinks only one channel needs to be tripped per the first section of note f.

B. Incorrect because the instrument being inop does not inop the EC, Plausible if the candidate goes directly to 3.1.3 and assumes 36-06A being inoperable makes both EC's inoperable. This is the action for both ECs inoperable.

C. Correct – 3.6.2c note f is applicable. The d/p instrument provides inputs to one instrument channel in each trip system. With less than minimum channels operable for both trip systems, place one in trip in one hour and the second one in trip within 24 hours.

D. Incorrect because the instrument being inop does not inop the EC, Plausible if the candidate goes directly to 3.1.3 and assumes 36-06A being inoperable makes the EC inoperable. This is the action for one EC inoperable.

References:	TS 3.1.3, 3.6.2, PID C-18017-C	Student Ref:	TS 3.1.3, 3.6.2, PID C-18017-C

Learning Objective:	N/A	
Question source:	NMP1	
Question History:	2002 NRC Exam	
Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X
10CFR	43.5	

QUESTION 88

A Plant startup is in progress. The Following conditions currently exist:

- All IRM Range switches are on Range 10
- The Mode Switch is in STARTUP

An event occurred which resulted in reduced recirculation pump flow:

- Total Core Flow is 19 Mlb/hr

Which one of the following states the Technical Specification requirements due to this condition, **and** the basis for this requirement?

	Tech Spec Requirement	Basis for TS Requirement
A.	The plant shall initiate a shutdown within 1 hour and be in cold shutdown within 10 hours.	To Prevent fuel cladding failure during a LOCA.
B.	Control rods shall not be Withdrawn.	To prevent exceeding the Safety Limit MCPR during a transient.
C.	The plant shall initiate a shutdown within 1 hour and be in cold shutdown within 10 hours	To Prevent fuel cladding temperature exceeding 1500 F during a LOCA.
D.	Control rods shall not be Withdrawn.	To prevent exceeding 1% plastic strain on the cladding during a transient.

K &A # 2.2.22

Importance Rating 4.7

QUESTION 88

K&A Statement: Knowledge of limiting conditions for operations and safety limits.

Justification:

A. Incorrect because the tech spec requirement is wrong and the basis is wrong. Plausible if the candidate can not find the correct tech spec. Almost all other NMP1 requirements are a 1 hour initiate shutdown and be in cold shutdown in 10 hours.

B. Correct – 20.25 is 30% flow per note in section 4 of N1-OP-43A. Per tech spec 3.1.7i control rods can not be withdrawn. The basis is to not exceed SLMCPR on a control rod withdrawal error to the full out position.

C. Incorrect because the tech spec requirement is wrong and the basis is wrong. Plausible if the candidate can not find the correct tech spec. Almost all other NMP1 requirements are a 1 hour initiate shutdown and be in cold shutdown in 10 hours.

D. Incorrect because the basis is not plastic strain. Plausible if the candidate does not know which limit is approached in a low flow condition in IRM range 10.

References: Tech Spec 3.1.7.i, Student Ref: TS 3.1.7

Learning Objective: N/A

Question source: Modified OC 08

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.2

QUESTION 89

Phase I core shuffle in progress to support Control Rod Blade and Drive maintenance and In Vessel inspections.

- SRM's 11, 12 and 14 are fully inserted and operable.
- The Reactor Mode Switch is in REFUEL
- Cell 30-13 is adjacent to SRM 12
- Cell 30-13 is disassembled and defueled in preparation for removal of Control Rod Blade (CRB) 30-13 and placement in the fuel pool.

Before CRB 30-13 is removed, the following events occur:

- SRM 14 is withdrawn during testing and counts drop to 90 cps.
- SRM 12 counts drop to 50 cps following defueling of cell 30-13

Which one of the following statements is correct in response to the above conditions?

A. The control rod blade may be moved into the fuel pool with the current condtions.

B. SRM 12 must be declared inoperable and CRB 30-13 may be placed in the fuel pool only after SRM 12 is made operable.

C. SRM 14 must be declared inoperable and CRB 30-13 may be placed in the fuel pool only after SRM 14 is made operable.

D. SRM 12 and SRM 14 must be declared inoperable and CRB 30-13 may be placed in the fuel pool only after SRM 12 AND SRM 14 are made operable.

K&A #	215004 A2.02
Importance Rating	3.7 (SRO)

QUESTION 89

K&A Statement: **A2.02** - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: **SRM inop condition**.

A. CORRECT A is correct – Per Tech Spec. 3.5.3, during major core alterations only two SRM's are required to be operable to support core alterations. SRM's are required to be operable in the quadrant where the core alteration is taking place and in one of the adjacent quadrants to the core alteration. Under the conditions presented, SRM 14 must be declared inoperable based on it being withdrawn from the core. To meet the specification 3.5.3.c requirement, the SRM's must be inserted to their normal operating level. SRM 12 is not required to be declared inoperable because T.S. 3.5.3 allows SRM count rates to drop to 3 cps during control rod drive maintenance as long as the SRM is fully inserted. Therefore with SRM 12 operable and SRM 11 operable in an adjacent quadrant the control rod blade may be moved to the fuel pool.

B. B is incorrect – SRM 12 is not required to be declared inoperable because T.S. 3.5.3 allows SRM count rates to drop to 3 cps during control rod drive maintenance as long as the SRM is fully inserted. This is a plausible distractor for those candidates that only review Tech Spec 3.5.1 and believe SRM 12 must be declared inoperable with counts below 100 cps.

C. C is incorrect – Under the conditions presented, SRM 14 must be declared inoperable based on it being withdrawn from the core. To meet the specification 3.5.3.c requirement, the SRM's must be inserted to their normal operating level. However, SRM 14 is not required to be operable to move the control rod blade per T.S. 3.5.3. This is a plausible distractor for those candidates that believe that 3 SRM's must be operable under all conditions and do not effectively review Tech Spec 3.5.3.

D. D is incorrect – As described above SRM12 remains operable and SRM 14 is inoperable but is not required for the performance of this particular core alteration.

References: N1-FHP-27C, Tech Spec 3.5.1, 3.5.3 Student Ref: Tech Spec's without setpoints,

Learning Objective: N/A

Question source: New

Question History: New

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 Comments: 41.10

QUESTION 90

The plant is operating at 95% power with four recirc pumps in service.

- Recirc loop 14 is shutdown with it's suction valve closed for scoop tube maintenance

At 22:30 the following occurs:

Recirc Pump #11 Trips due to an electrical fault

What are the required actions from this event?

A. Reactor power must be monitored and maintained less than 90% until the pump can be isolated. No Tech Spec ACTIONS are required.

B. Enter an immediate shutdown and manual scram to prevent exceeding MCPR safety limit.

C. Within 1 hour, SCRAM the Reactor or close the Recirc pump discharge valve.

D. Within 12 hours, SCRAM the Reactor or close the Recirc pump discharge valve.

K&A #	215005 A2.07
Importance Rating	3.4 (SRO)

QUESTION 90

K&A Statement: **A2.07**-Ability to (a) predict the impacts of the following on the APRM / LPRM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **Recirc flow channels flow mismatch.**

A. A is incorrect – Reactor power must be reduced below 90% power as a required action in N1-SOP-1.3 when the Recirc pump discharge valve fails to close. This condition was not provided and is not required in this case and although Reactor power is expected to be below 90% with two pumps out of service it is not the required action under these conditions. Additionally Tech Spec. 3.6.2 is applicable due to the trip of the recirc pump making the flow biased scram and rod block setpoints non-conservative. This is a plausible distractor for those candidates that do not recognize the APRM inoperability due to the recirc pump trip.

B. B is incorrect – An immediate SCRAM is required when there are less than 3 Recirc pumps operating per N1-SOP-1.3. This operating condition is prohibited. Since there are still three pumps in operation this action is not required. Additionally, those candidates that immediately enter the Otherwise statement in Action (o) will determine that the ACTION in 3.6.2a(1) must be taken which is to insert control rods. This is a plausible distractor for those candidates that misinterpret the requirement to scram as three recirc pumps versus less than three recirc pumps or misinterpret the interpretation of the ACTION required under Item (o).

C. CORRECT C is correct – With the current conditions all four APRM's are inoperable due to their flow biased SCRAM and Rod Block Setpoints being non-conservative due to reverse flow through the recirc loop causing artificially high total flow readings. Declaring all four APRMs inoperable requires actions under Tech Spec 3.6.2a(1) and 3.6.2a(7). With less than the required number of APRMs, Note (o) is entered and the first item requires "Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter ...". This one hour window is the only time allowable because in the current condition there are no channels available for the Parameter which requires entry into the "Otherwise" statement at the bottom of Note (o). The Otherwise statement requires the ACTION in 3.6.2a to be taken. The ACTION under Item 7 is to not withdraw rods which is not limiting here. The ACTION under Item 1 is to insert control rods. Since this is not a shutdown action it can only be accomplished by a Reactor SCRAM. Therefore there is one hour to close the Recirc discharge valve and return the APRMs to an Operable status.

D. D is incorrect – As described above the APRMs are inoperable and Tech Spec 3.6.2a is required to be entered. Those candidates that misinterpret Item 1 of Note (o) and believe they can utilize the * Note at the bottom of Note (o) will determine that the trip does not need to be inserted on either channel because it would cause the trip to occur. This would allow 12 hours to restore one of the channels to an operable status prior to taking the ACTION under 3.6.2a(1).

References: N1-SOP-1.3, Tech Spec 3.6.2a

Student Ref: Tech Spec's without setpoints,

Learning Objective: N/A

Question source: New

Question History: New

Cognitive level:	Memory/Fundamental knowledge:	
	Comprehensive/Analysis:	X

10CFR 55 41.5 45.6

Comments:

QUESTION 91

The plant is operating at 100% power when the following alarms are received:

- F1 2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO
- F1 2-2, MAIN STEAM RAD MONITOR CH 12 HI/LO

Main Steam line radiation readings and setpoints are:

- 900 mR/hr (setpoint)
- 3100 mR/hr (indicated)

Two minutes later the following alarm is received:

- H1-1-7, OFF GAS HIGH RADIATION alarm

Chemistry samples have confirmed the HIGH RADIATION alarm with the following results:

- 136 mR (setpoint)
- 750 cps (sample results / indicated)

Which one of the following describes the required operator action(s) for the high radiation conditions that exist?

A. Declare an ALERT and enter N1-EOP-6, Radioactivity Release Control

B. Declare an UNUSUAL EVENT and enter N1-EOP-5, Secondary Containment Control

C. Declare an UNUSUAL EVENT and reduce reactor power per N1-SOP-1.1 as necessary to control radiation levels

D. Declare an ALERT, SCRAM the reactor per N1-SOP-1 and initiate Manual Vessel Isolation per N1-SOP-40.2

K&A # 271000 A2.04
Importance Rating 4.1 (SRO)

QUESTION 91

K&A Statement: A2.04 – Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: **Offgas System high radiation**.

A. A is incorrect – The current Offgas Radiation conditions do not warrant an ALERT declaration and the subsequent entry into EOP-6. This is a plausible distractor for those candidates that miscalculate the Offgas release rate and determine that the current conditions are 10x the alarm setpoint.

B. B is incorrect – An unusual Event is required to be declared based on the current Offgas Rad conditions. Entry into EOP-5 is not required because the Main Steam Line or Offgas Rad monitors do not meet the entry conditions for EOP-5. This is a plausible distractor for those candidates that incorrectly assess the MSL or Offgas process radiation monitors as part of the EOP-5 entry conditions.

C. CORRECT C is correct – Based on the current Offgas Radiation levels at 5.5x the alarm setpoint an Unusual Event is the correct classification per EAL 1.1. Additionally, the Offgas radiation condition requires power to be reduced to control radiation level which progresses into a shutdown N1-OP-43C if radiation conditions can not be corrected. The current MSL radiation level at 3.0 times the alarm setpoint does not require additional actions per N1-SOP-25.2 or the EAL Matrix.

D. D is incorrect – As described above an alert declaration on offgas radiation is not warranted for the current conditions. Additionally, the current Main Steam Radiation Levels do not require a Reactor SCRAM and manual vessel isolation. This is a plausible distractor for those candidates that miscalculate the offgas and / or Main Steam Line Radiation conditions.

References: ARP H1 1-7, N1-SOP-25.2, N1-EOP-5, N1-EOP-6 Student Ref: EAL Matrix

Learning Objective: N/A

Question source: NMP1 2000 NRC Exam

Question History: NMP1 2000 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5 45.6

Comments:

QUESTION 92

- A refueling outage is ongoing when the Control Room Air Treatment System is declared inoperable.

With current plant conditions, which of the following Operations with the Potential to Drain the Vessel (OPDRV) may continue without additional approvals?

A. CRD Mechanism removal and replacement per the approved vendor procedure. Core Spray is inoperable for the duration of the replacements.

B. Replacement of a section of the 2" RWCU Bottom Head drain piping in between the reactor and the first manual isolation valve using a freeze seal. Core Spray is inoperable for the duration of the replacement.

C. Repairs to the #14 Recirc Pump Discharge which requires inserting an air bladder and maintaining drains open on the discharge of the pump due to isolation valve leak-by. Core Spray is operable for the duration of the repair.

D. All of the described activities can continue with Control Room Air Treatment system inoperable.

K&A #	G2.1.40
Importance Rating	3.9 (SRO)

QUESTION 92
K&A Statement: **G2.1.40** – Knowledge of refueling administrative requirements

A. CORRECT A is correct – Per N1-OP-34, an Operation with the Potential to Drain the Vessel (OPDRV) is defined as an activity; with the potential to uncover irradiated fuel. To preclude a maintenance or operational activity on piping to the reactor connecting below the top of active fuel from being classified as an OPDRV, at least one of the following conditions must be satisfied:

1. A manual or automatic valve shut in line with the activity (effectively isolating it).
2. A flange in line with the activity (effectively isolating it).
3. Other device in line with the activity APPROVED by Engineering
4. Activity on a line or penetration that is < 6 inch in diameter and Core spray is operable per Technical Specifications.
5. Activity on a line or penetration with Engineering Supporting Document concluding that makeup flow available is greater than leakage flow thru the line.
6. Control Rod Drive Mechanism removals / replacements per Vendor Procedure UV-BWR-001, CRD Exchange Using the Model 11 Exchange Machine.

Since CRD removals and replacements are a covered exception to the definition of an OPDRV, this activity may continue with the CREVS system inoperable.

B. B is incorrect – The replacement of the piping does not meet the exclusion criteria 3, 4 or 5 since there is no engineering analysis supporting the freeze seal as an equivalent isolation device or Engineering Support document that available make-up is greater than the leakage through the 2" line. With Core Spray inoperable, this activity also does not meet the exclusion for lines less than 6" in diameter. This is a plausible distractor for those candidates that believe the exclusion criteria apply due to the size of the piping.

C. C is incorrect – The repair on the 28" discharge line does not meet the exclusion criteria for work on a line <6" and Core Spray being operable. This is a plausible distractor for those candidates that do not recognize the limitation in pipe size that comes with Core Spray operability in order to meet the OPDRV exclusion criteria.

D. D is incorrect – Only the CRD maintenance is excluded per the definition provided in N1-OP-34. This is a plausible distractor for those candidates that confuse the Tech Spec requirements for CREVS versus RBEVS. If RBEVS were inop then OPDRVs could continue for seven days before they would need to be suspended.

References: N1-OP-34, C-18006-C, C-18009-C Student Ref: None

Learning Objective: N/A

Question source: New

Question History: New

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.5 45.6
Comments:

QUESTION 93

Core Shuffle Part II is in progress. The following two moves have been made.

1. A double blade guide (DBG) was lifted in the Spent Fuel Pool. All indications on the Refuel Bridge were correct for lifting the DBG The DBG was transferred and released in the core.

2. A fuel bundle was lifted in the Spent Fuel Pool. All indications on the Refuel Bridge were correct for lifting the fuel bundle. The fuel bundle was transferred over the reactor core.

- During both moves, the following indications were observed to be in the following state the entire time of the moves:

In the Control Room the Rod Block Monitor Panel REFUEL INTERLOCK indicator light is NOT illuminated.

On the Refuel Bridge the ROD BLOCK INTERLOCK Light on the refueling bridge is NOT illuminated.

What refueling interlock/limit switch has failed and when should have the interlock/limit switch been in effect?

	Interlock/Limit Switch Failure	**Interlock/Limit Switch in Effect**
A.	The refuel bridge one rod out interlock failed.	When the fuel bundle was over the core.
B.	The refuel bridge one rod out interlock failed.	When the DBG was over the core.
C.	The "over core" limit switch on the refuel bridge.	When the fuel bundle was over failed the core.
D.	The "over core" limit switch failed on the refuel bridge.	When the DBG was over the core.

K&A #	234000 K4.02
Importance Rating	K4.02 (SRO) 4.1

QUESTION 93

K&A Statement: **K4.02 -** Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: **Prevention of control rod movement during core alterations**

A. A is incorrect – The refuel bridge one rod out interlock would generate a bridge reverse motion stop, fuel hoist motion block, and rod block would be received, not just a rod block. This is a plausible distractor for those candidates that do not recognize that even if the one rod out interlock had failed, with the "over core" limit switch functional the ROD BLOCK and other normal indications would be received.

B. B is incorrect – The refuel bridge one rod out interlock would generate a bridge reverse motion stop, fuel hoist motion block, and rod block would be received, not just a rod block. A failure of this interlock would only be recognized with a fuel bundle loaded on the hoist since the double blade guide does not weigh enough to meet the hoist loading requirement to complete the refuel interlock. This is a plausible distractor for those candidates that do not recognize that even if the one rod out interlock had failed, with the "over core" limit switch functional the ROD BLOCK and other normal indications would be received.

C. CORRECT C is correct – The over-the-core limit switch has failed which prevents a rod out block from being generated when the bridge is loaded and over the reactor core. The failure can only be recognized after the hoist has fuel loaded and the refuel bridge is over the core. At that point the REFUEL INTERLOCK indicator light on the Rod Block Monitor Panel and the ROD BLOCK INTERLOCK Light on the refueling bridge will light and annunciator F3-4-4, ROD BLOCK should alarm.

D. D is incorrect – The "over core" limit switch has failed; however the double blade guide does not weigh enough to meet the hoist loading requirement to complete the refuel interlock and create the Rod block. This is a plausible distractor for those candidates that are unsure of the double blade guide weight or unaware of the weight requirement in the refuel interlock.

References: N1-ST-W3, N1-FHP-25, N1-OP-34 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC Exam

Question History: NMP1 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5 45.6

Comments:

QUESTION 94

The plant is starting up after an outage.

The following chemistry samples have been taken over the past three days at 0800 at each day.

	Day 1	Day 2	Day 3
Reactor coolant pH	8.9	9.1	9.1
Reactor coolant conductivity(μmho/cm)	1.19	.57	.19
Reactor coolant chlorides (ppm)	175	110	25
Reactor coolant sulfates (ppb)	25	20	18
Steam flow (lb/hr)	50,000	2,000,000	4,500,000

Which one of the following is required IAW Technical Specifications and at what time should the action statement have been entered if required(based upon the above data)?

A. No Technical Specifications have been exceeded. No actions required.

B. Shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.
Enter the action statement at 0800 on the first day.

C. Shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.
Enter the action statement at 0800 on the second day.

D. Shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.
Enter the action statement at 0800 on the third day.

K&A # G2.1.34
Importance Rating 3.5

QUESTION 94

K&A Statement: Ability to maintain primary and secondary plant chemistry within allowable limits.

Justification:

A. Incorrect because tech spec limits are exceeded. Plausible if the candidate is confused by the various requirements for time, power level and chemistry parameters.

B. Incorrect because tech spec 3.2.3.a would only be applicable if above the limit for > 24 hours. Since this is the first sample 24 hours have not passed. Plausible if the candidate ses that the limit is exceeded but does not realize it can be exceeded for 24 hours.

C. Correct – Tech spec 3.2.3.c.3 requires with thermal power >10% the maximum limit of chlorides is 100 ppb. This has been exceeded on the second day.

D. Incorrect because tech spec 3.2.3.b has been exceeded, however 3.2.3.c.3 was exceeded 24 hours earlier. Plausible if the candidate sees that the limit is exceeded but does not realize that 3.2.3.c.3 was exceeded previously.

References: Tech Spec 3.2.3

Student Ref: Tech Spec 3.2.3

Learning Objective: N/A

Question source: New

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.5

QUESTION 95

The phase I core shuffle is in progress.

Which one of the following conditions meets the Fuel Handling Procedure (FHP) criteria for stopping fuel movement?

A. Last performance of the Refueling Platform Interlocks Test was completed twenty-four (24) hours ago.

B. The fuel assembly nose piece is lowered to two (2) feet above the core top guide before establishing the correct orientation.

C. There is a loss of an offsite power supply, but the remaining offsite power is above the Technical Specification minimum requirement.

D. There is a loss of rod position indication for a control rod that is in a different core quadrant from where the fuel moves are being performed.

K&A # G2.1.35
Importance Rating 3.9

QUESTION 95
K&A Statement: Knowledge of the fuel-handling responsibilities of SROs.
Justification:

A. Incorrect because This interval for this surveillance is 7 days. The surveillance is current. Plausible because a similar surveillance N1-PM-SO, Refuel Platform and Grapple Inspection, is required to be completed every 12 hours.

B. Incorrect because it is acceptable to lower the fuel assembly before establishing the correct orientation, however, the correct orientation is to be established before lowering the fuel assembly into the assigned core location. Plausible if the candidate thinks the correct orientation must be established prior to lowering the bundles.

C. Incorrect N1-FHP -25 attachment 4 requires to stop moves with a loss of the minimum offsite power. Fuel moves do not need to be stopped if the site meets the tech spec required offsite power. Plausible because the procedure does mention offsite power.

D. Correct – The loss of positive indication that all control rods are fully inserted is a reason to stop fuel movements. It does not matter which quadrant of the core the loss of indication occurs.

References: N1-FHP-25 Student Ref: None

Learning Objective: N/A

Question source: Modified NMP1 2005 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.7

QUESTION 96

A plant startup at the Beginning of the Cycle is in progress, with the following:

- Reactor power is 85%
- 13 Feedwater pump is the only feedpump in service
- Core Flow is 77%
- Recirc Flow control is in automatic
- Tau =.8
- MCPR is determined to be 1.50

Which one of the following describes when the first action is required?

A. 15 Minutes

B. 1 Hour

C. 2 Hours

D. No action is required

K&A # G2.2.39
Importance Rating 4.5

QUESTION 96

K&A Statement: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Justification:

A. Correct –MCPR limit with given conditions would be (from figure) 1.47 X (Kf from figure 2e) 1.10=1.62. 1.62>1.50 and therefore TS 3.1.7c is applicable. Action shall be initiated within 15 minutes to restore to within limits.

B. Incorrect but plausible if the candidate thinks that they are into a shutdown situation, NMP1 requires reducing power within 1 hour.

C. Incorrect but plausible if the candidate takes the second action of 3.1.7c. Which is if not returned within 2 hours power reductions shall be initiated.

D. Incorrect but plausible if the candidate does not use Kf to adjust MCPR. Then not in 3.1.7c action because MCPR limit would be 1.47.

References: Tech Spec 3.1.7c, COLR

Student Ref: Tech Spec 3.1.7c COLR

Learning Objective: N/A

Question source: NMP1

Question History: 2002 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 43.2

QUESTION 97

A Type 1 Procedure Change Evaluation (PCE) was generated for a Unit One (1) Operating procedure. The Technical Verifier, QTR Review, and RPO Review are complete.

Per N1P-PRO-04, Procedure Change Evaluations and Future Procedure Enhancements, which one of the following “approval” satisfies ALL approval requirements to consider this PCE complete?

A. WEC SRO approves the PCE before performing it AND CRS approves the PCE within 14 days of the WEC SRO approval. No other approval is required to consider this PCE complete.

B. SM or CRS approves the PCE before implementing it. No other approval is required to consider this PCE complete.

C. Manager Operations or GSO approves the PCE within 14 days of its performance. No other approval is required to consider this PCE complete.

D. CRS approves the PCE before implementing it AND GSO approves the PCE within 14 days of the CRS approval. No other approval is required to consider this PCE complete.

K&A # G2.2.6
Importance Rating 3.6

QUESTION 97
K&A Statement: Knowledge of the process for making changes in procedures.
Justification:

A. Incorrect because although the WEC is an SRO the CRS can not sign for the manager or director or GS required review. Plausible if the candidate thinks that 2 SRO reviews are required for procedure changes.

B. Incorrect because after SRO approval all type 1 procedures need manager, director or GS approval within 14 days. Plausible if the candidate does not know of the additional 14 day review period. -

C. Incorrect because all Type 1 procedures need an SRO review. Plausible if a Type 2 procedure change that does not require SRO review.

D. Correct – Per procedure need a SRO review and either a manager director or GS to review and approve within 14 days.

References: N1P-PRO-04 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC

Question History: None

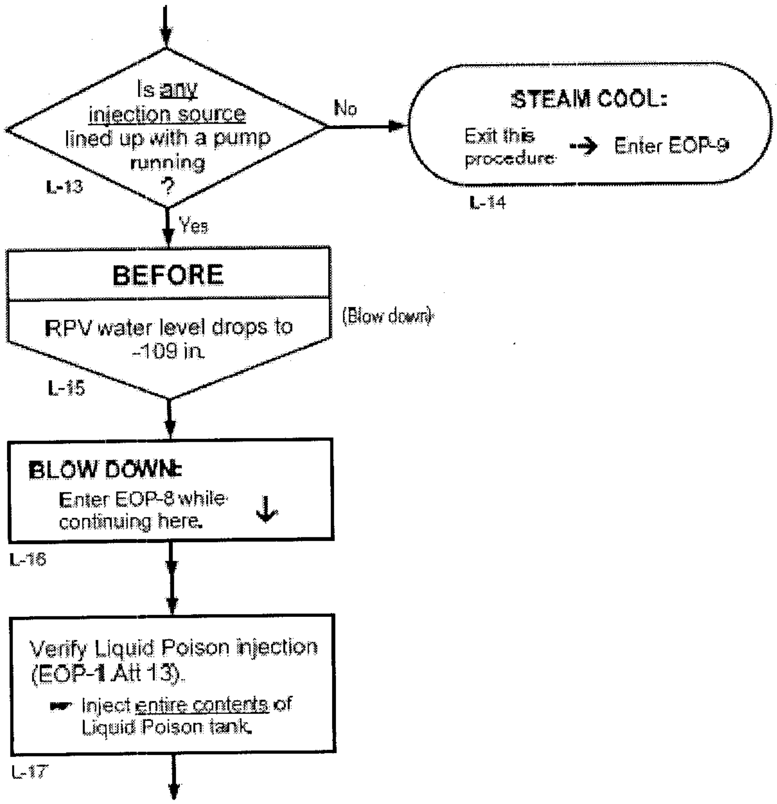
Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.3

QUESTION 98

A loss of coolant accident has occurred.

EOP-2 has been entered with reactor level continuing to drop.



What is the basis for injecting Liquid Poison per Step L-17 of EOP-2?

A. Provide an additional source of injection to the reactor.

B. Minimizes radioactive release by making the suppression pool pH higher (less acidic).

C. Minimizes radioactive release by making the suppression pool pH lower (more acidic).

D. Minimizes radioactive release by borating the reactor coolant inside the reactor vessel.

K&A # G2.3.14
Importance Rating 3.8 (SRO)

QUESTION 98

K&A Statement: **G2.3.14 -** Knowledge of the radiation or contamination hazards that may arise during normal, abnormal or emergency conditions or activities.

A. A is incorrect – Establishing Liquid Poison in Step L-17 is not as a means for level control but as a means to control radiological dose following a lossof coolant accident involving core damage. Since Liquid Poison is identified as an Alternate Injection System it would likely be started to augment RPV injection in an earlier step of the Level branch, before RPV water level reaches the top of the active fuel (Element L-3, L-7, or L-9). This is a plausible distractor for those candidates that do not recognize the radiological impact from Liquid Poison injection once the TAF has been reached.

B. CORRECT B is correct – Design basis analyses credit Liquid Poison injection for limiting the radiological dose following loss of coolant accidents involving core damage. Radiation induced reactions are predicted to convert large fractions of dissolved ionic iodine into elemental iodine and organic iodides which can escape into the containment atmosphere. The rate of these reactions is strongly dependent on suppression pool pH. If the pH is maintained greater than 7, very little of the dissolved iodine will be converted to volatile forms and most of the iodine fission products will be retained in the suppression pool. Over time, the pH in the torus will tend to decrease due to the addition of acidic chemicals. The sodium pentaborate solution used in the Liquid Poison system is derived from a strong base and therefore raises suppression pool pH.

C. C is incorrect – As described above, Liquid Poison is injected to control and raise Torus pH following the onset of a LOCA. This is a plausible distractor for those candidates who are unsure of the addition of Liquid Poison raises or lowers pH in the Torus.

D. D is incorrect – Boration of the reactor coolant is performed to reduce power levels in the core by neutron moderation. This is plausible distractor for those candidates who believe that dose mitigation is achieved with boration of the coolant in the vessel versus the torus volume.

References: SAP-1, NER-1M-095, REV 2 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC Exam

Question History: NMP1 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.5 45.6
Comments:

QUESTION 99

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

00:10 Plant conditions justify the declaration of an ALERT
00:17 Plant conditions change and only meet the UNUSUAL EVENT threshold.

Which one of the following describes the EPIP-EPP-20 emergency notification to be made in response to these conditions and the latest time at which the notification must be made?

A. 00:25 Declare and report the UNUSUAL EVENT. The transmittal form should indicate the ALERT conditions that occurred.

B. 00:32 Declare and report an UNUSUAL EVENT. Include conditions justifying an ALERT existed momentarily but cleared.

C. 00:25 Declare and report an ALERT. Include conditions were momentary and current plant conditions only justify an UNUSUAL EVENT.

D. 00:32 Declare and report an ALERT. Include conditions were momentary and current plant conditions only justify an UNUSUAL EVENT.

K&A #	G2.4.40
Importance Rating	4.5 (SRO)

QUESTION 99

K&A Statement: **G2.4.40** - Knowledge of SROs responsibilities in emergency plan implementation.

A. CORRECT A is correct – Per the requirements of EPIP-EPP-20 and EPIP-EPP-01, when conditions exist for a declaration but then change and a different classification should be made based on current conditions, then the notification is made at the current (even if potentially lower) classification level. The timeliness of the notification is based on the initial entry into the emergency condition. The change in conditions does not eliminate the requirement of the 15 minute notification from the indication that an EAL was exceeded.

B. B is incorrect – Although the notification should be made as described, the timeliness of the notification is incorrect based on the discussion in answer A. This is a plausible distractor for those candidates that believe the time clock as well as the notification level change with the changing conditions.

C. C is incorrect – The notification level here is incorrect for a transitory event. Declaration at the highest level achieved is not warranted in this case. The timeliness of the declaration is correct for this answer. This is a plausible distractor for those candidates that do not recognize that this notification may be made utilizing the transitory event requirements of the referenced procedures.

D. D is incorrect – The notification level here is incorrect for a transitory event. Declaration at the highest level achieved is not warranted in this case. The timeliness of the declaration is also not correct. This is a plausible distractor for those candidates who incorrectly determine that the change in plant conditions allows the notification to be made at the later time using the higher classification.

References: EPIP-EPP-01 3.1.4, EPIP-EPP-20 Sect. 3.2, EPIP-EPP-25 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC Exam

Question History: NMP1 2004 NRC Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55 41.10 43.5 / 45.11

Comments:

QUESTION 100

The reactor was initially operating at rated power. A feedwater line break inside primary containment resulted in a high drywell pressure scram. Current plant conditions are as follows:

- RPV level is 88 inches and rising slowly with both CRD pumps injecting
- RPV pressure is being maintained at 800-900 psig with Isolation Condensers
- Drywell pressure is 13 psig and lowering slowly with containment sprays initiated
- Containment Spray total flow is 1.6 mlbm/hr from all running pumps
- Torus water temperature is 112 F and lowering slowly

Torus water level is 161.5 inches and rising slowly

Step TL-4 of EOP-4 directs to maintain torus water level below 13.5 feet.

What is the basis for maintaining torus level below 13.5 feet?

A. The Torus boundary design load would be exceeded if ERV's were opened with the Torus level above 13.5 feet.

B. Maintaining torus level below 13.5 feet ensures that the torus has the heat capacity to sustain an RPV blowdown.

C. Maintaining torus level below 13.5 feet ensures that the torus pressure will not exceed the Primary Containment Pressure Limit during an RPV blowdown.

D. The pressure suppression of the primary containment is assumed to function as designed only when primary containment water level is below 13.5 feet.

K&A # G2.4.18
Importance Rating 4.0

QUESTION 100
K&A Statement: Knowledge of the specific bases for EOP's.
Justification:

A. A is incorrect –The torus boundary design limit is a combination of torus pressure and the hydrostatic pressure from the torus water level. This is not limiting at NMP1 and is not the basis for maintaining level below 13.5 feet. Plausible because the torus boundary design limit is in the basis and is dependant on torus level.

B. B is incorrect – The heat capacity temperature limit is derived based on torus levels, having more water will increase the heat capacity. If the HCTL is being approached the strategy would be to lower pressure to ensure there is adequate heat capacity. Using EOP-1 Attachment 23 the current parameters will come close to the HCTL curve when reactor pressure is at or around 900 psig but margin will be gained with the reduced Torus temperature. Plausible because the stem has the HCTL being approached and the candidate may think that the high torus level limit is based on not having enough heat capacity.

C. C is incorrect – The Primary Containment Pressure Limit is a function of Torus level, however 100.5 feet is the limit due to being the elevation of the highest containment vent. Higher levels than 100.5 feet are not permitted because the containment could not be then vented.

D. CORRECT D is correct – The Maximum Pressure Suppression Primary Containment Water Level is the highest primary containment water level at which the pressure suppression capability sufficient to accommodate an RPV breach by core debris can be maintained. For Mark I containments, the Maximum Pressure Suppression Primary Containment Water Level is defined to be the bottom of the downcomer ring header. Irrespective of the torus airspace volume, the pressure suppression feature of the primary containment can be assumed to function as designed only when primary containment water level is below this elevation. From EOP basis.

References: N1-EOP-4 basis Student Ref: None

Learning Objective: N/A

Question source: Modified OC 2006 Exam

Question History: None

Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 55 41.10/43.1