KEY
NMP1 2009 NRC Written Exam

1	B	35	A	69	D
2	D	36	C	70	B
3	A	37	C	71	D
4	B	38	B	72	C
5	C	39	A	73	B
6	D	40	A	74	C
7	C	41	C	75	D
8	B	42	A	76	A
9	C	43	B	77	B
10	C	44	A	78	A
11	D	45	B	79	D
12	D	46	B	80	A
13	C	47	A	81	B
14	D	48	D	82	C
15	B	49	D	83	B
16	D	50	B	84	A
17	D	51	D	85	B
18	D	52	A	86	B
19	D	53	D	87	C
20	B	54	B	88	B
21	C	55	C	89	A
22	C	56	C	90	C
23	D	57	D	91	C
24	C	58	C	92	A
25	B	59	C	93	C
26	A	60	C	94	C
27	A	61	B	95	D
28	C	62	D	96	A
29	D	63	A	97	A
30	A	64	D	98	B
31	B	65	A	99	A
32	B	66	C	100	D
33	A	67	C		
34	B	68	C		

QUESTION 1

The plant is operating at 50% power with 4 Recirc Pumps in operation, when the following occur:

- A malfunction in the master recirculation controller causes recirculation flow and reactor power to lower
- The Reactor Operator places all individual recirculation M/A stations in MANUAL and the flow/power reduction ceases
- The following conditions result:
    - Reactor power is 30% and steady
    - Reactor recirculation flow is 20% of rated core flow and steady

Which one of the following actions is required?

A. Manually Scram the Reactor.

B. Raise recirculation flow or insert rods.

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

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

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 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.

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

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

QUESTION 2

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

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

Then, a fault occurs on Powerboard 12 with the following:

- Supply breakers R122 and R123 trip
- Operators perform a manual scram

Which one of the following describes the reason for the manual scram?

A. Service Water pumps are not available.

B. Only two Reactor Recirc Pumps are running.

C. Only three Reactor Recirc Pumps are running.

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

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

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. Correct – #11 Condenser Circulating Water Pump is tagged OOS and the loss of PB 12 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.

References: N1-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

Which one of the following is the reason for performing Battery Load reductions during a station blackout?

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

B. Avoid a loss of critical battery board loads due to over current or overload.

C. Maintain power to the Process Computer and Annunciators for the entire coping period.

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. Incorrrect. Plausible if the applicant thinks that as loads draw more current they may be lost due to overload or overcurrent.

C. Incorrect. Plausible if the applicant does not know what loads are shed. PPC and annunciators are de-energized when MG 167 is tripped 2 hours into the 4 hour coping period.

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 plant is operating at 35% power with the following:

- The turbine generator is on-line
- The Reactor Operator is raising reactor power by withdrawing control rods
- Then, turbine vibrations slowly rise to 15 mils and stabilize

Which one of the following describes the required Operator action?

A. Immediately scram the Reactor.

B. Immediately trip the Turbine. A Reactor scram is not required.

C. Scram the Reactor if vibrations are not lowered within 15 minutes.

D. Trip the Turbine if vibrations are not lowered within 15 minutes. A Reactor scram is not required.

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 if the candidate does not recognize that at 35% power, first stage bowl pressure is low enough to bypass Reactor scrams from the Turbine trip

B. Correct – ARP A2-3-5 requires the turbine to be tripped immediately if vibrations exceed 12 mils. With power at 35%, first stage bowl pressure is low enough to bypass Reactor scrams from the Turbine trip, therefore a scram is not required.

C. Incorrect but plausible if the candidate does not recognize that 15 mils is above the 10-12 mil range where a 15 minute clock applies prior to tripping the turbine.

D. Incorrect but plausible if the candidate does not recognize that 15 mils is above the 10-12 mil range where a 15 minute clock applies prior to tripping the turbine, and does not recognize that at 35% power, first stage bowl pressure is low enough to bypass Reactor scrams from the Turbine trip.

References: ARP A2-3-5 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

A plant startup is in progress with the following:

- Reactor power at 10% and stable
- The mode switch is in STARTUP
- Recirculation flow is 60% of rated core flow
- Reactor pressure is 1000 psig and stable

Then, a turbine bypass valve malfunction causes the following:

- Reactor pressure spikes to 1063 psig
- Reactor power spikes to 40%

Which one of the following describes 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 1080 psig.

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 60% rated flow the APRM flow biased scram setpoint would be in excess of 90% 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 evacuated due to a fire with the following:

- The immediate actions of N1-SOP-21.2, Control Room Evacuation, have been taken
- An operator is stationed at Remote Shutdown Panel 11
- The Channel 11 Control Transfer keylock switch has been taken to the EMERG position

Which one of the following describes the impact of this switch manipulation on the Emergency Condenser controls?

The high pressure Emergency Condenser initiation _______(1)_______. This action ______(2)_____ Emergency Condenser Isolation.

A. (1) Has been disabled
(2) Will prevent

B. (1) Is still functional
(2) Will prevent

C. (1) Has been disabled
(2) Will **NOT** prevent

D. (1) Is still functional
(2) Will **NOT** 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, some aspects of the control room Emergency Condenser controls are disabled in EMERG, however the auto initation signals are still available. Also the isolation is not prevented.

B. Incorrect but plausible, it does not prevent auto isolation.

C. Incorrect but plausible, some aspects of the control room Emergency Condenser controls are disabled in EMERG, however the auto initation signals are still available.

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

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 the following:

- One Reactor Building Closed Loop Cooling (RBCLC) pump is tagged out of service.
- An electrical problem causes one of the running RBCLC pumps to trip.
- Annunciator H1-4-1, R Building Cooling Water Press Temp Makeup Flow, is in alarm.
- RBCLC supply temperature is 95°F and rising slowly.
- N1-SOP-11.1, RBCLC Failure, has been entered.

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

A. Drywell Air Coolers.

B. Fuel Pool Heat Exchangers.

C. Reactor Water Cleanup Pumps.

D. Air Compressor 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. 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.

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

C. 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.

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

Which one of the following air compressors is available to supply instrument air?

A. Service Air Compressor

B. Instrument Air Compressor 11

C. Instrument Air Compressor 12

D. Instrument Air Compressor 13

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 IAC 12 can only be powered from EDG 103 which failed to start.

D. 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.

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 is shutting down for an outage, with the following conditions:

- RPV coolant temperature is 195°F and slowly lowering.
- Shutdown Cooling pumps 11 and 12 are in service with a total SDC system flow of 4000 gpm.
- RPV water level is being maintained at 73 inches indicated level.

Then, all of the Reactor Recirculation pumps trip. Shutdown Cooling flow has been verified at 4000 gpm.

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

A. Maximize SDC flow with SDC pumps 11 and 12.

B. Maximize SDC flow by placing SDC pump 13 in service.

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

D. Raise reactor level above the main steam line nozzles and shut all recirc pump discharge 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, 3 SDC pump operation needs vessel level to be at the reactor flange for NPSH requirements.

C. 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.

D. Incorrect but plausible if the applicant does not realize that by leaving the recirc suction and discharge bypass valves open, the SDC flow will still bypass the reactor.

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 dropped on the floor or storage vault per N1-SOP-34, however the bundle is dropped in the SFP. Dropping a bundle in the SFP requires the refuel floor and drywell to be evacuated.

B. Incorrect but plausible if the applicant does not know that N1-SOP-34 requires CREVS initiation.

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

A small LOCA occurs in the drywell, resulting in the following sequence of events:

Time	Event
02:18	Venting primary containment through RBEVS initiated
02:47	The reactor is manually scrammed
02:51	Drywell pressure peaked at 3.8 psig
03:03	RPV water level dropped to a low of +15 inches and stabilized
03:10	Drywell pressure lowered to 3.0 psig and stabilized
03:23	Condenser Vacuum is 17" Hg and lowering

Given the following systems:

1. Reactor Water Cleanup System
2. Reactor Coolant Sampling System
3. Containment H2-O2 Monitoring System

Which one of the following identifies the above system that is isolated and requires the isolation signal to be bypassed or reset in order to place the system in service, if any?

A. None

B. Reactor Water Cleanup System

C. Reactor Coolant Sampling System

D. Containment H2-O2 Monitoring System

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 exceeded 3.5 psig.

B. B 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.

C. C 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.

D. CORRECT D is correct - Containment isolation has occurred with drywell pressure above 3.5 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. Alternately, the high DW pressure signal can be reset.

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 is operating at 100% power when EPR sensed pressure fails downscale.

Which one of the following describes the plant response to this malfunction?

A. The Reactor scrams on high RPV pressure.

B. The Reactor scrams on a low pressure MSIV isolation.

C. The Reactor remains at power. The MPR controls pressure at a new, lower value.

D. The Reactor remains at power. The MPR controls pressure at a new, higher value.

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, plausible if the candidate does not understand that the pressure regulator with the higher error signal controls the TCVs, and therefore the MPR will take over control.

B. Incorrect, plausible if the candidate does not understand that the low sensed pressure will result in the EPR throttling the TCVs closed.

C. Incorrect, plausible if the candidate does not understand that the low sensed pressure will result in the EPR throttling the TCVs closed.

D. Correct – The EPR will attempt to close TCVs, resulting in rising RPV pressure. As MPR pressure error becomes greater than EPR pressure error, the MPR will assume control of TCVs, limiting the pressure rise.

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

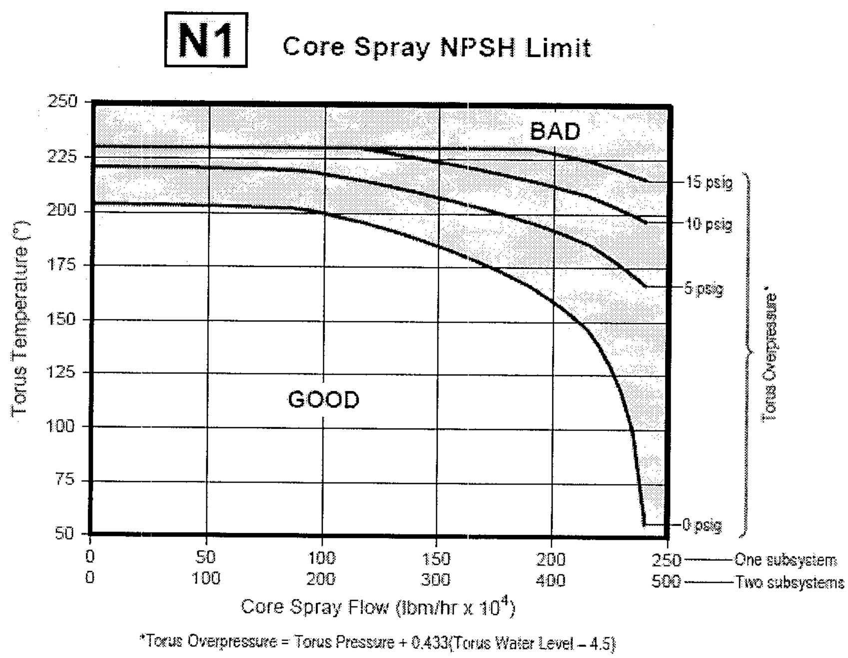
10CFR 41.7

QUESTION 13

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

- The RPV has been depressurized using ECs and ERVs
- Core Spray is injecting and maintaining Reactor level
- Containment Sprays have been utilized to lower Containment pressure
- Torus water temperature is 200°F
- Torus level is 9.5 feet
- Torus pressure is 6 psig
- Drywell pressure 7 psig
- An electrical fault has resulted in the following:
    - 40-12, Core Spray Discharge IV 11 (Outside), is failed closed
    - 40-01, Core Spray Discharge IV 121 (Inside), is failed closed

Which one of the following states the maximum Core Spray flow (lbm/hr x 10^4) that may be used for RPV injection?



A. 175

B. 240

C. 350

D. 480

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 calculates the overpressure correctly but does not recognize that two subsystems of Core Spray are available, at 200F in the Torus this would yield a one subsystem flow of approximately 175. 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 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.

C. Correct C 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 results in injection from Core Spray Loop 11 being unavailable. This leaves two subsystems (pumps) in Loop 12 available for injection. The failure of 40-01 does not effect the number of available subsystems. With two subsystems available the max flow is 350.

D. D 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.

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

Learning Objective: N/A

Question source: NMP1 2002 NRC Exam

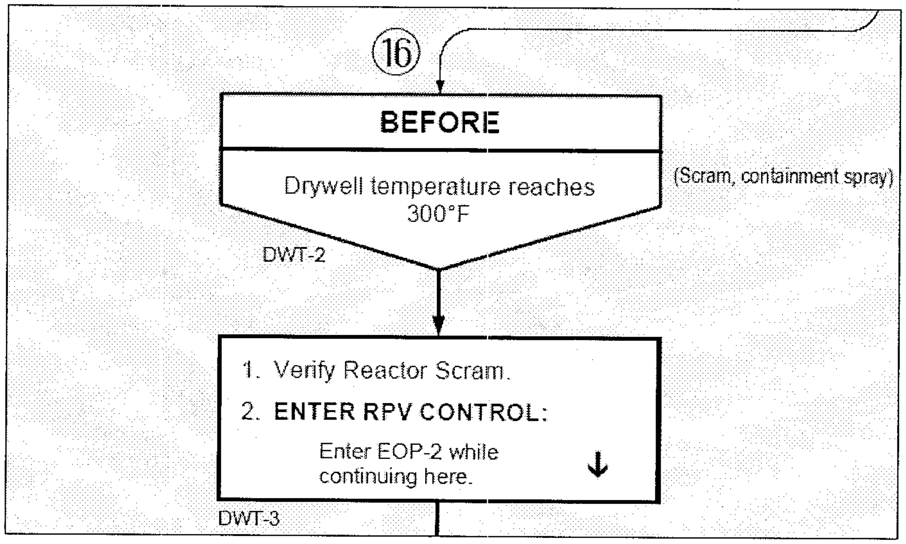
Question History: NMP1 2002 NRC Exam

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

10CFR 55 41.8

QUESTION 14

Given the following steps from N1-EOP-4, Primary Containment Control:



Which one of the following describes the reason for scramming the Reactor in step DWT-3?

A. This prevents exceeding the qualification of the Reactor vessel level and pressure instrumentation cabling.

B. This ensures that Drywell Sprays will be effective.

C. This prevents RPV water level inaccuracies.

D. This reduces the heat input to the Drywell.

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. A 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.

B. B 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.

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.

CORRECT D 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.

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

The plant is operating at 100% power when a Main Turbine trip and an ATWS occur, resulting in the following:

- Reactor power is 6%.
- Reactor water level has been lowered in accordance with N1-EOP-3, Failure to SCRAM.
- Reactor pressure is being controlled between 800 and 1000 psig with Emergency Condensers and ERVs.
- Torus water temperature is 117°F.
- Torus water level is 10.6 feet and lowering slowly.

Assuming all parameters remain on their current trend, which one of the following describes the impact on the current Reactor pressure control band?

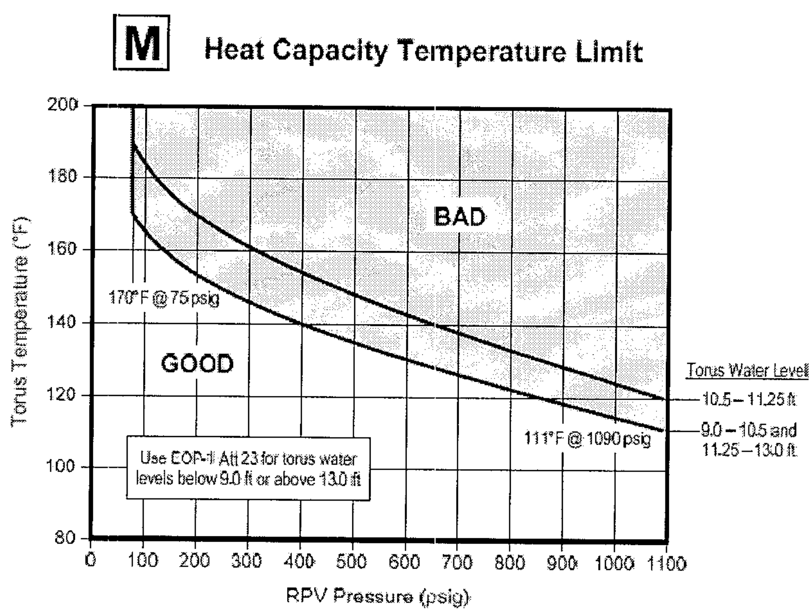


Figure 3-11: Heat Capacity Temperature Limit

A. An emergency blowdown is required

B. Reactor pressure must be maintained at or below 900 psig

C. Reactor pressure must be maintained at or above 900 psig

D. Continued use of the entire current 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 900 psig) will result in exceeding the HCTL Limit as Torus level continues to drop. 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 drop. Torus level is approaching the limit to transition to 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: None

Learning Objective: N/A

Question source: Modified PB 2005 Exam

Question History: None

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

10CFR 55 41.10

QUESTION 16

A plant transient has resulted in the following:

- Reactor pressure is 150 psig and steady
- No ERVs are open
- RPV water level is -118" and steady
- Core Spray pumps 111 and 112 have tripped
- Core Spray loop 12 is injecting at 190×10^4 lbm/hr

Which one of the following describes the status of core cooling?

A. Spray Cooling ensures adequate core cooling

B. Steam Cooling with injection ensures adequate core cooling

C. Steam Cooling without injection ensures adequate core cooling

D. There is no assurance of 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. A 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. This requirement is included in Step L-18 in EOP-2 as a limitation prior to entering the SAPs.

B. B 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.

C. C 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.

D. CORRECT D 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.

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

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

- An ATWS has occurred
- N1-EOP-3 is entered
- ARI is initiated
- No rods have inserted

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 425 psig
- NO CRD pumps are available

Which one of the following methods in N1-EOP-3.1 could be used to insert control rods?

A. Increasing Cooling Water differential pressure

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 – Increasing the cooling water differential pressure in order to drive rods in requires operation of at least one CRD pump to be effective. With no CRD pumps in operation this method of rod insertion is not available. This is a plausible distractor for those candidates that do not recognize that the stem identifies both CRD pumps are out of service.

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

An unisolable steam leak from the Main Steam lines into the Turbine Building has occurred.

N1-EOP-6, Radioactive Release Control, has been entered and contains a step to operate Turbine Building ventilation, as required.

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

A. Prevent unmonitored ground level releases.

B. Provide a controlled and elevated release point.

C. Maintain access to the Turbine Building for damage control team actions.

D. Minimize dose impact on feedwater HPCI components required for level control.

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. B 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.

C. C 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.

D. CORRECT D 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. Operation of the system is not credited for HPCI or feedwater operation and the components are not environmentally qualified. This is a plausible distractor because the ventilation system does maintain the environment in the turbine building but is not part of an environmental qualification of the HPCI components and is not described in the bases step of EOP-6 describing why Turbine Building ventilation is put in service.

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

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

- Smoke Purge is running and aligned to the Auxiliary Control Room for fire surveillance testing.
- Then, a fire occurs in the Auxiliary Control Room.
- Fire detection for the Auxiliary Control Room actuates.
- The Auxiliary Control Room high exhaust duct temperature alarm is received at Local Fire Panel 1.
- A small amount of smoke has also entered the Main Control Room.

Which one of the following describes the response of the Smoke Purge system and the ability to operate Smoke Purge for the Main and Auxiliary Control Rooms?

A. Smoke Purge system flow continues throughout the fire and will clear smoke from the Auxiliary Control Room with no further operator action.

B. Smoke Purge system flow continues throughout the fire and will clear smoke from the Main and Auxiliary Control Room with no further operator action.

C. Smoke Purge system isolates. Following reset, the Smoke Purge system can be aligned to both the Main and Auxiliary Control Room simultaneously.

D. Smoke Purge system isolates. Following reset, the Smoke Purge system can be aligned to only the Main Control Room or the Auxiliary Control Room at one time.

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. Incorrect – The high duct temperature condition isolates the smoke purge system. Plausible if the candidate does not know that smoke purge automatically isolates on high duct temperature.

B. Incorrect – The high duct temperature condition isolates the smoke purge system. Plausible if the candidate does not know that smoke purge automatically isolates on high duct temperature. The aux and main control rooms must be purged separately. Plausible if the candidate does not understand the interlock preventing simultaneous operation.

C. Incorrect – The aux and main control rooms must be purged separately. Plausible if the candidate does not understand the interlock preventing simultaneous operation.

D. D is correct – Upon receipt of the high duct temperature, the smoke purge system will isolate. An interlock prevents simultaneous operation of aux and main control room smoke purge to prevent smoke from traversing from one area to the other.

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

Learning Objective: N/A

Question source: New

Question History: None

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

10CFR 55 41.7

QUESTION 20

The plant is operating at 100% power during degraded grid conditions.

Main Generator conditions are as follows:

- 640 MWe
- 200 MVAR
- H_2 Pressure is 43 psig

Then, the following alarms occur:

- A1-4-1, GENERATOR H2 SEAL OIL PRESSURE LOW, annunciates
- A1-4-2, GENERATOR HYDROGEN SYSTEM, annunciates
- Computer point D113 GEN H2 SYS SUP OIL PR _ LOW has been received
- Seal oil differential pressure at the Generator is 0.5 psid
- Generator H_2 pressure is 28 psig and stable
- The Main Seal Oil Pump (MSOP) is still in service
- MSOP discharge pressure is 105 psig

Note: The Generator Estimated Capability curve is provided on the following page.

Which one of the following describes the first action required to be taken under these conditions?

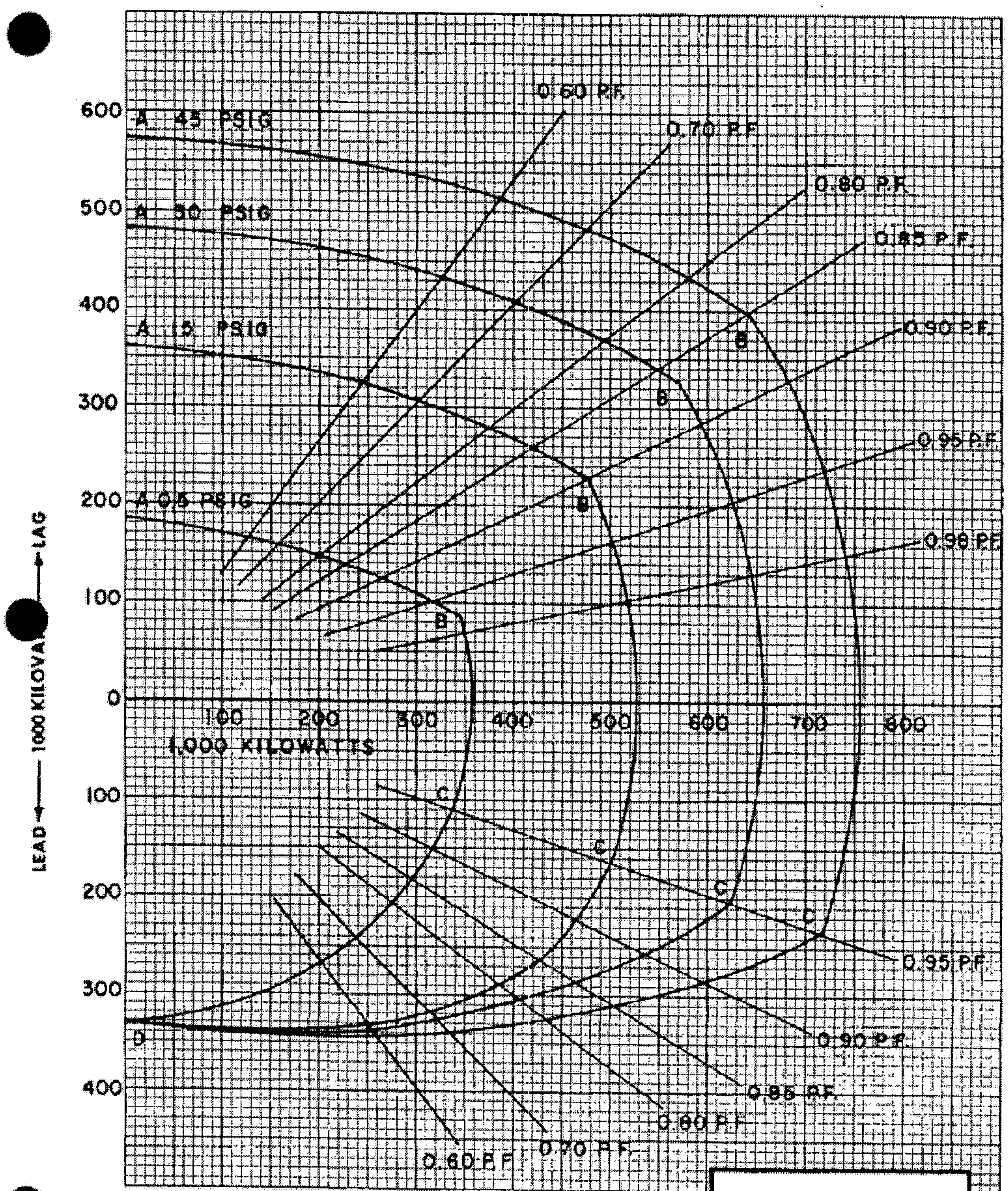
A. Add H_2 to the Generator per N1-OP-7

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

C. Start the Emergency Seal Oil Pump and secure the MSOP per N1-SOP-32

D. Scram the Reactor per N1-SOP-1 and verify the Turbine trip per N1-SOP-31.1

ATB 4 POLE, 755,000 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

Estimated capability curves
Dwg. 326HA727 (rev 0)

Fig. 17-3

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 – 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.

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 – The given MSOP discharge pressure indicates the pump is operating properly. Securing the pump is therefore not a requirement. SOP-32 requires verification that MSOP or ESOP are operating, not both. Plausible if the candidate misdiagnoses the MSOP discharge pressure as too low (ESOP should start if MSOP discharge pressure drops below 90 psig).

D. D 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.

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

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

- Six drywell cooling fans are running
- RBCLC pumps 11 and 12 are running

Then the following events occur:

- The feeder breaker to PB161A 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?

A. RBCLC flow to the DW has isolated

B. A RBCLC pump has lost power, only

C. Three DW air coolers have lost power, only

D. A RBCLC pump and three DW air coolers have lost power

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

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

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

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:

- Annunciator L1-4-4, Drywell-Torus Temp High, is in alarm
- Drywell Temp Avg CH 11 is 140 degrees and rising slowly
- Drywell Temp CH 12 Elev 319 is in alarm at 180 degrees and rising slowly
- Drywell Temp CH 11 Elev 230 is in alarm at 140 degrees and rising slowly
- Drywell pressure is 1.2 psig and rising slowly
- Drywell humidity is 20% and slowly lowering
- Drywell unidentified leakage is 0.22 gpm and steady
- Six Drywell Fans are in service
- Service Water is valved into 11 RBCLC Heat Exchanger

Which one of the following describes the required operator action?

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

B. Reduce Reactor power per N1-SOP-1.1, Emergency Power Reduction

C. Take manual control of 70-23B, RBCLC TCV, and lower RBCLC temperature.

D. Obtain Shift Manager permission to bypass alarming points. Increase frequency of monitoring of Drywell temperatures.

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

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

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

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

The plant is operating at 100% power.

Which one of the following malfunctions will result in a stable higher power level once steady state conditions are achieved?

A. Main Condenser Circulating Water Pump trip.

B. Inadvertent start of the Standby Feedwater Booster Pump.

C. Inadvertently isolating the Reactor Water Cleanup System.

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

B. Incorrect because the transient level rise will result in a momentary thermal power rise, but the feedwater level control system will recover level and core thermal power will return to the pre-transient level.

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

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

Which one of the following explains the reason for intentionally lowering RPV water level during an ATWS?

A. Concentrate boron in the core

B. Lower void fraction inside the shroud

C. Raise preheating of the incoming feedwater

D. Raise natural circulation through the core to mix boron

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 in the core will change due to the change in core inlet subcooling, and boiling boundary will lower.

C. Correct –Per EOP 3 Failure to scram basis, Element L-5 override states that uncovering the sparger heats the incoming feedwater, thereby reducing the subcooling at the core inlet. This leads to lower power and reduced risk of power oscillations.

D. Incorrec. Plausible because once hot shutdown boron is injected, core flow is raised to provide adequate mixing of boron into the core region.

References: EOP-3 Basis

Student Ref: None

Learning Objective: N/A

Question source: LIM 08 NRC Exam

Question History: None

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

10CFR 55.41

QUESTION 25

The plant is starting up following a refueling outage, with the following sequence of events:

Time	Event
09:27	Preparations are in progress for starting the first Feedwater pump.
09:30	Annunciators F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB, and F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarm.
09:32	CRD Pump 12 does not start when its control switch is placed to START.
09:34	Annunciator F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, alarms. Control Rod 10-19 accumulator light is lit and accumulator pressure is reported as 900 psig and lowering.
09:37	Control Rod 34-35 accumulator light is lit and accumulator pressure is reported as 920 psig and lowering.

Which one of the following lists the time at which a manual scram is first required?

A. 09:32

B. 09:34

C. 09:52

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 who believe that as soon as both pumps are non-operational the plant must be scrammed.

B. CORRECT B 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 with no CRD pumps running and receipt of the 1st accumulator trouble alarm.

C. C is incorrect – This is a plausible distractor for those candidates who believe that as soon as both pumps are non-operational, and with a RPV pressure above 900#, a 20 minute clock must be started.

D. D is incorrect – This is a plausible distractor for those candidates who do not understand that the initial Feedwater pump start during a plant startup is conducted well below 900 psig.

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 fails 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 and computer points are received in the Control Room:

- H1-4-8, Area Radiation Monitors, alarms
    - F306 RB340 REFUEL LO RNG is displaying 650 MR/H
    - F307 RB340 REFUEL HI RNG is displaying 650 MR/H
    - F316 RB340 EAST WALL is displaying 25 MR/H

- L1-4-3, React Bldg Vent Rad Monitor Off Normal, alarms
    - E476 RB VENT RMON 11 is displaying 3.8 MR/H
    - E477 RB VENT RMON 12 is displaying 6.0 MR/H
    - F307 RB340 REFUEL HI RNG is displaying 650 MR/H

Which one of the following describes the automatic ventilation system response?

A. Reactor Building Ventilation trips due to RB Vent Rad Monitor #12. RBEVS initiates and aligns to the Reactor Building.

B. Reactor Building Ventilation trips due to Refuel Bridge High Range Rad Monitor. RBEVS initiates and aligns to the Reactor Building.

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

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

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

QUESTION 26

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 a Containment Spray Room water level rising above the Maximum Safe Value. The CRS directs a manual Reactor scram per N1-EOP-5.

Which one of the following describes the basis for inserting the scram?

A. Reduces RPV pressure to decrease flow through the break.

B. Ensures reactor is shutdown prior to leak getting larger.

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

D. Allows personnel into secondary containment to perform safe shutdown actions.

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. CORRECT A 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

The basis to SCRAM the reactor for a "primary system" leak is driven by the reduction in flow through the leak from a drop in RPV pressure. From the EOP bases, "A system is a "primary system" if it is connected directly to the RPV and if reducing RPV pressure will decrease flow through a break in the system."

B. B 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.

C. C 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.

D. D 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.

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 with the following conditions:

- Two SDC pumps are in operation
- Two RBCLC heat exchangers are in service
- Five Recirculation pumps are in operation
- Recirculation loop temperatures are 330°F and stable
- Reactor water level is +60 inches and stable
- SDC flow is 4000 gpm and stable

Which one of the following describes a problem with the current conditions?

A. SDC has failed to isolate on high temperature.

B. Total SDC flow is too low to ensure adequate core circulation.

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

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 pumps trip at 350 F. Plausible because an isolation of the system would stop decay heat removal.

B. 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.

C. Correct – With 2 pumps running the SDC pumps need >65 inches for NPSH, unless coolant temperature is below 150°F.

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 when a main generator fault causes a turbine trip and Reactor scram. The following conditions then occur:

- RPV pressure is 918 psig and steady.
- RPV water level is 65 inches and steady.

Which one of the following failures would result in this steady-state RPV water level?

A. #13 Feedwater Pump Flow instrument has failed high.

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

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

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

K&A # 206000 K3.01
Importance Rating 4.0

QUESTION 29

K&A Statement: **Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following:** Reactor Water Level Control.

Justification:

A. 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..

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

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

D. 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.

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

The plant 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.
- Feedwater pumps 11 and 12 have tripped on high level.
- Reactor water level is 96" and lowering slowly.
- Reactor pressure is 920 psig and being controlled automatically by the EPR.

Which one of the following describes the response of the Feedwater system?

A. Feedwater pumps 11 and 12 will automatically start and inject when Reactor water level reaches 53".

B. Feedwater pumps 11 and 12 will automatically start and inject when Reactor water level reaches 90".

C. Feedwater pumps 11 and 12 will not restart automatically until the operator manually resets the high level trip.

D. Feedwater pump 12 will automatically start and inject when Reactor Level reaches 53". Feedwater pump 11 will NOT automatically start.

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 plant was operating at 100% power when the following occurred:

- MSIVs closed and the Reactor scrammed
- Emergency Cooling System initiated
- Emergency Cooling Loop 11 auto isolated
- Then, Emergency Cooling Channel 11 bypass switch is placed in the BYPASS position

Which one of the following describes the position of the Emergency Cooling (EC) system valves if Reactor pressure now exceeds 1100 psig for 20 seconds?

	EC Condensate Return Valve 11	EC Steam Isolation Valve 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 experienced a small break LOCA with the following:

- Plant has scrammed
- Drywell pressure is 4.5 psig and rising slowly
- Reactor water level is -20 inches and lowering slowly
- Channel 11 ADS RELAY and CONFIRM white lights are NOT illuminated
- Channel 12 ADS RELAY and CONFIRM white lights are illuminated

Which one of the following describes the status of Core Spray and the ERVs ~~two~~ TEN minutes later?

All Core Spray pumps are running…

A. On minimum flow to the Torus.

B. And after 111 seconds, the primary ERVs opened allowing Core Spray to inject into the vessel.

C. And after 115.5 seconds, the backup ERVs opened allowing Core Spray to inject into the vessel.

D. And after 115.5 seconds, the primary and backup ERVs opened allowing Core Spray to inject into the vessel.

NOTE: REVISED DURING EXAM ADMINISTRATION. - LICENSEE REQUESTED CHANGE IN RESPONSE TO APPLICANT'S QUESTION. THE CHANGE ENSURED THAT ANSWER CHOICE "B" WAS THE ONLY CORRECT ANSWER. THIS ALLOWED ADEQUATE TIME FOR THE ERV'S OPEN & TO BLEED RX PRESSURE DOWN BELOW THE CORE SPRAY LP INTERLOCK TO ALLOW INJECTION

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 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.

B. 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.

C. 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.

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 the following:

- A fault has caused a loss of RPS Bus 11
- Reactor pressure is 305 psig and lowering slowly
- Drywell pressure is 3.7 psig and rising slowly

Which one of the following describes how Core Spray will respond to these conditions?

A. All Core Spray pumps will start.

B. 111 and 112 Core Spray and Core Spray Topping Pumps will start, only.

C. 121 and 122 Core Spray and Core Spray Topping Pumps will start, only.

D. No Core Spray pumps will start.

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 12 Core spray loop lost initiation logic.

C. Incorrect - because the failure will not prevent an initiation signal. Plausible if the candidate thinks that the 11 Core spray loop lost initiation logic.

D. 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.

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 when a Containment isolation resulted in the following:

- Reactor Power is 15%
- ERVs are cycling to control reactor pressure
- Liquid Poison Pump has been manually started from the Control Room

Which one of the following describes indications that the squibs have fired and the explosive valves are open?

A. Current meter on Panel 1S-65 in Auxiliary Control Room Indicates 0.15 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 with the following:

- MODE switch is in STARTUP.
- All IRMs are on Range 8.
- All MSIVs are open.
- Reactor Pressure is 200 psig.

Then, the operator moves one RPS Channel 11 IRM to Range 10 and one RPS Channel 12 IRM to Range 10.

Which one of the following describes the plant response and the reason for this response?

All MSIVs...

A. Close and the Reactor scrams because of the MSIV closure scram.

B. Close, however the Reactor remains operating because the MSIV closure scram is bypassed when less than 600 psig.

C. Remain open because the low pressure closure of the MSIVs is bypassed in the STARTUP mode.

D. Remain open because the low pressure closure of the MSIVs is bypassed by the combination of the MODE switch in STARTUP and the current IRM Range switch positions.

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. 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.

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 the bypass is only mode switch dependant.

D. 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.

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

Which one of the following describes how RPS functions to insert control rods?

A. Backup scram valve solenoids de-energize, which then vents instrument air pressure through the scram pilot valves to atmosphere.

B. Backup scram valve solenoids energize, which then vents instrument air pressure through the backup scram valves to atmosphere.

C. Scram pilot valve solenoids de-energize, which then vents instrument air pressure through the scram pilot valve solenoids to atmosphere.

D. Scram pilot valve solenoids energize, which then vents instrument air pressure through the scram pilot valve solenoids to atmosphere.

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 both backup scram valve solenoids have to de-energize to scram rods, not energized.

C. 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.

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

References: Student Ref: None

Learning Objective: N/A

Question source: Pilgrim 2007 NRC exam

Question History: None

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

10CFR 41.7

QUESTION 37

The plant startup is in progress with the following:

- The Mode Switch is in STARTUP with control rod withdrawals in progress
- IRMs 11, 12, 15, 16, and 18 read 72-74 on Range 2
- IRMs 13, 14, and 17 read 9-10 on Range 3

Then, a malfunction in the IRM drive circuitry causes IRM 13 to withdraw to the full-out position.

Which one 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 rod block from IRM downscale **ONLY**; bypassing the IRM is required to continue withdrawing control rods.

C. This will result in panel annunciators and a rod block 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 rod block 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 Instrumentation Bus 11.

Which one of the following describes the response of the Nuclear Instruments?

	SRMs	IRMs	APRMs
A.	No change	Four Fail Low	No change
B.	Two Fail Low	Four Fail Low	No change
C.	Two Fail Low	Four Fail Low	Four Fail Low
D.	Two Fail 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-19866-C shts 1, 2, 3, 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: X
Comprehensive/Analysis:

10CFR 41.7

QUESTION 39

A plant startup is in progress with the following:

- The reactor is critical and a heatup is in progress
- SRMs are partially withdrawn
- 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 switch is in Non-Coincident
- Refuel Instrument Trip Bypass 12 switch is in Non-Coincident
- Electrical power is lost to SRM 13 high voltage power supply

Which one of the following describes the effect of this loss and the necessary action(s) to continue withdrawing control rods?

A. Rod Block due to loss of voltage. Bypass SRM 13 and continue withdrawing rods.

B. SRM 13 fails downscale. Control rod withdrawal can continue with no further action.

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

D. Rod Block due to loss of voltage. Rod withdrawals must be suspended 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 SRM would need to be bypassed to continue the startup, but plausible because the indication would fail downscale without detector voltage. If the candidate does not know that a rod block would come in then they 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 light on the Flow Control Trip Reference (FCTR) card for APRM 12 is lit solid red
- APRM 12 INOP light is lit
- The OATC places APRM 12 joystick to bypass
- Then, 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 both APRM 12 and APRM 18

B. A ½ scram will occur on RPS channel 12 only because APRM 18 is failed upscale

C. A ½ scram will occur on RPS channel 12 only and a Rod Block will occur because APRM 18 is failed upscale

D. A ½ scram will occur on RPS channel 12 only because APRM 18 is failed upscale and a Rod Block will occur because APRM 12 is inoperable

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 bypassing 12 will bypass the ½ scram for RPS 11.

C. 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.

D. 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.

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

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

- A simultaneous loss of Powerboard 102 and Battery Board 11 occurs

Which one of the following identifies the status of the ADS Logic Channels and ERVs that can actuate on an ADS signal?

	Available ADS Logic Channels	ERVs Available for ADS
A.	Channel 11 only	Primary Valves
B.	Neither Channel	Primary Valves
C.	Channel 12 only	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. 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. 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. 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. 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:	ADS Lesson Plan, N1-OP-45, C-19859-C	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:

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

Which one of the following describes the status of Reactor Water Cleanup (RWCU)?

A. RWCU system isolates. RWCU can NOT be used for pressure control per N1-EOP-1 Attachment 9, RPV Press Control thru RWCU Temperature.

B. RWCU system isolates. RWCU can be used for pressure control per N1-EOP-1 Attachment 9, RPV Press Control thru RWCU Temperature.

C. RWCU system does not isolate. RWCU can be used for pressure control per N1-EOP-1 Attachment 9, RPV Press Control thru RWCU Temperature.

D. RWCU system does not isolate. RWCU can be used for pressure control per N1-EOP-1 Attachment 9, RPV Press Control thru RWCU Temperature, or 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 de-energized
- A sustained loss of Turbine Building ventilation occurs
- MSIVs isolate on high steam tunnel temperature
- Reactor Pressure peaks at 1240 psig

Which one of the following describes the pressure control mechanisms that actuated and the status of the Reactor Pressure Safety Limit during the transient?

	Pressure Control Mechanisms	Reactor Pressure Safety Limit
A.	Only Safety Valves opened	Exceeded
B.	ERVs and Safety Valves opened	NOT exceeded
C.	Only ERVs opened	Exceeded
D.	No ERVs or Safety Valves opened	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 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 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 RPV water level control strategy and the reason for this strategy?

A. Control RPV water level low on GEMAC level instruments to avoid high level trips from Yarway level instruments.

B. Control RPV water level high on GEMAC level instruments to avoid low level trips from Yarway level instruments.

C. Control RPV water level low on Yarway level instruments to avoid high level trips from GEMAC level instruments.

D. Control RPV water level high on Yarway level instruments to avoid low level trips from GEMAC level instruments.

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 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:

- Annunciator L1-4-3, REACT BLDG VENT RAD MONITOR OFF NORMAL is lit
- Reactor Building Exhaust Radiation Monitor 12 indicates downscale

Five minutes later the following occurs:

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

Which one of the following describes the response of RBEVS?

A. Neither Reactor Building Ventilation Radiation channel generates a trip signal and RBEVS does NOT start.

B. Only Reactor Building Ventilation Radiation channel 11 generates a trip signal and RBEVS starts.

C. Only Reactor Building Ventilation Radiation channel 12 generates a trip signal and RBEVS starts.

D. Both Reactor Building Ventilation Radiation channels 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 Ventilation Radiation channel 11 to 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 Ventilation Radiation channel 11. 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 Ventilation Radiation channel 11 to 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 channel 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 Ventilation Radiation channel 11 to 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 channels 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 with the Main Generator supplying 630 MWe and 100 MVAR to the grid.

Which one of the following describes the Main Generator response to placing the voltage regulator control switch to RAISE (assume an infinite grid)?

A. Generator speed will rise

B. Generator field voltage will rise

C. Generator power factor will rise

D. Generator output voltage will rise

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. CORRECT B 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".

C. C is incorrect – The initial conditions place the generator supplying VARs to the grid which is a lagging power factor. From these conditions raising field voltage to the generator will cause VAR to increase and real power (MWe) to remain the same. This will cause power factor to lower. If the generator had been in a leading power factor this answer would have been correct.

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 100% power with the following:

- PB 102 Emergency Bus Under Voltage relay 27-1 fails its surveillance and is placed in the tripped condition
- During repairs, PB 102 Emergency Bus Degraded Voltage relay 27-1A also trips
- All meter indications of PB 102 voltage are normal in the main control room

Which one of the following describes the response of EDG 102 during these events?

EDG 102...

A. Remains in standby.

B. Fast starts when bus under voltage relay 27-1 is tripped, and the output breaker remains open.

C. Fast starts when bus degraded voltage relay 27-1A trips, and the output breaker closes.

D. Fast starts when bus degraded voltage relay 27-1A trips, and the output breaker remains open.

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 27-1A similar to the actions taken for 27-1.

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 degraded voltage 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, C-19410-C sht 11 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 the following:

- 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.

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

RPS Bus 11 is...

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

B. De-energized and must be manually transferred to I&C Bus 130A.

C. Powered by the bypass transformer with 125 VDC from Battery Board 11 with no loss of power.

D. Powered by the bypass transformer 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. 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.

C. 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.

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 when the following alarm is received:

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

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

A. Static Battery Charger 171 maintains DC power to Battery Board 12 without a loss of power.

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

C. MG Set 167 automatically swaps to “BATTERY CHARGE” mode and supplies Battery Board 12 after a momentary 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

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

- An automatic start signal is received on EDG 102
- 5 seconds after start signal the engine speed reaches a peak of 175 rpm
- Annunciator A4-3-5, DSL. GEN. 102 START-RUN OFF NORMAL, alarms

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

A. The EDG 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 EDG 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 EDG 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 EDG 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 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. 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

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

- Frequency - 60 Hz
- Real load - 100 KW
- Reactive load - 60 KVAR

Which one of the following actions will establish correct operating conditions per N1-ST-M4B?

A. Place DIESEL GOV control switch to RAISE to establish VARs to the generator.

B. Place DIESEL GOV control switch to RAISE to establish VARs to the bus.

C. Place VOLT ADJ RHEO GEN 103 to RAISE to establish VARs to the generator.

D. Place VOLT ADJ RHEO GEN 103 to RAISE to establish VARs to the bus.

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 change 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. C is incorrect – Placing the voltage rheostat to RAISE will increase the VARs the diesel is providing to the bus. This is a plausible distractor because the candidate must recognize the correct switch to be used and know that the diesel supplies kVAR to the bus vice being accepted by the generator during normal operation.

D. CORRECT D is correct - Per N1-ST-M4B Step 8.1.16, "If performing N1-ST-M4B 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 vars to the bus. 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.

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 operating at 100% power with the following:

- Instrument Air Compressors (IAC) 11 and 13 are in service
- IAC 12 control switch is tagged in pull-to-lock for maintenance
- Instrument Air Dryer (IAD) 94-168 is in service
- Subsequently, IAC 11 trips and its control switch is placed in pull-to-lock
- The CRS directs bypass of IAD 94-168 and IAD 94-169

Which one of the following actions is required until either IAC 11 or IAC 12 is returned to service, in accordance with N1-OP-20?

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 at 100% power with the following:

- 11 TBCLC pump is running
- 12 and 13 TBCLC heat exchangers are in service
- Condenser inlet temperature is 45°F
- An air line break causes the control air pressure to drop to 0 psig from TBCLC temperature controller TC-71-20 to both of the following valves:
  - 71-88, TBCLC TCV
  - 72-147, Service Water (SW) to TBCLC Drag Valve

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 72-93R, SW Drag Valve Bypass
B.	Rise	Manually control 71-88, TBCLC TCV, and 72-147, SW to TBCLC Drag Valve
C.	Lower	Throttle 72-93R, SW Drag Valve Bypass
D.	Lower	Manually control 71-88, TBCLC TCV, and 72-147, SW to TBCLC Drag Valve

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-24, 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 the following:

- Both the Reactor Building Inner Lift Door and the Outer Swing Door are closed.
- An operator is tasked with opening the Outer Swing Door.
- Prior to opening the Outer Swing Door, the operator verifies closed the Inner Lift Door, and notes that the Seal Pressure is 0 psig.

Which one of the following describes the impact of this Seal Pressure on Secondary Containment and the required actions?

A. Secondary Containment is met and the Outer Swing Door may be opened.

B. Secondary Containment is met but the Outer Swing Door must be maintained closed.

C. Secondary Containment is NOT met and the Outer Swing Door must be maintained closed.

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

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. Plausible if the applicant does not think that the inner seal being at 0 psig would make the door inop and therefore can not open the door.

B. Correct – Secondary Containemnt is met since the outer door is closed, however with the seal not inflated per N1-OP-52 step F.1.3.1, the outer door cannot be opened.

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. 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.

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. Unaffected due to Recirc Loop 11 being available.

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 SDC does not utilize Recirc Loop 11. This is plausible because Recirc Loop 11 has a system connection with EC11.

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

QUESTION 56

A plant 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 16 and 24, respectively
- All Group 7 Rods, except two, are at 12
- The remaining two Group 7 rods are at 04
- All Group 8 Rods are at 16

Then, the following events occur:

- A Group 8 Rod is selected with the intention to withdraw the rod to position 24.
- The operator goes the wrong direction and inadvertently gives the rod a continuous insert signal.

Which one of the following describes the response of the rod, and at what point a RWM rod block will be generated, if any?

A. The rod will insert to 00. No Rod Blocks are generated.

B. The rod will insert to 14 and an insert error Rod Block will be generated. The rod will settle at 14.

C. The rod will insert to 12 and an insert error Rod Block will be generated. The rod will settle at 12.

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

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 does not realize that this rod is the 3rd insert error which will generate a rod block.

B. Incorrect. Plausible if the applicant thinks the error will happen at 14 which is beyond the insert limit. However at 14 the alternate limit is satisfied and there will not be a rod block.

C. 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.

D. 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.

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 plant is operating at 100% power.

Which one of the following corresponds to an EOP entry condition AT THIS TIME, assuming that each of the given alarms has just actuated?

A. K2-4-3, Drywell Pressure High-Low, is in alarm. Drywell pressure is rising slowly.

B. F2-3-3, React Vessel Level High-Low, is in alarm. Reactor water level is lowering slowly.

C. F2-3-4, React Vessel Pressure High, is in alarm. Reactor pressure is rising slowly.

D. K1-1-1, Rx Bldg Area Temp High, is in alarm. Emergency Condenser area temperature computer points are in alarm and rising slowly.

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 this is below the EOP entry condition of 3.5 psig (3.2 psig EOP entry annunciator). Plausible if the candidate does not know this condition is below the EOP entry condition.

B. Incorrect because this is above the EOP entry condition of 53 inches (55 inch EOP entry annunciator). Plausible if the candidate does not know this condition is above the EOP entry condition.

C. Incorrect because this is below the EOP entry condition of 1080 psig (1068 psig EOP entry annunciator). Plausible if the candidate does not know this condition is below the EOP entry condition.

D. Correct – Reactor Building Area High Temperature alarm, if valid is an EOP entry condition for EOP-5. The presence of the corresponding computer points validates the alarm.

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

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

- Torus temperature is 80°F and rising slowly due to a leaking ERV.
- Then, Powerboard 102 trips on a ground fault.

Which one of the following describes the Containment Spray loops available for Torus Cooling?

A. Loops 111 and 112

B. Loops 112 and 122

C. Loops 121 and 122

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

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

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

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 one of the following describes the normal makeup water flow path to the Spent Fuel 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 an MSIV closure time limit and the associated basis for the stated limit?

	MSIV Closure Time Limit	Basis for MSIV Closure Time Limit
A.	3 Seconds	In conjunction with Main Steam Line flow limiters, ensures core coverage following a Main Steam Line rupture.
B.	3 Seconds	In conjunction with MSIV position scram, ensures ERV actuation is not required following Main Steam Line isolation at power.
C.	10 Seconds	In conjunction with Main Steam Line flow limiters, ensures core coverage following a Main Steam Line rupture.
D.	10 Seconds	In conjunction with MSIV position scram, ensures ERV actuation is not required following Main Steam Line isolation at power.

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

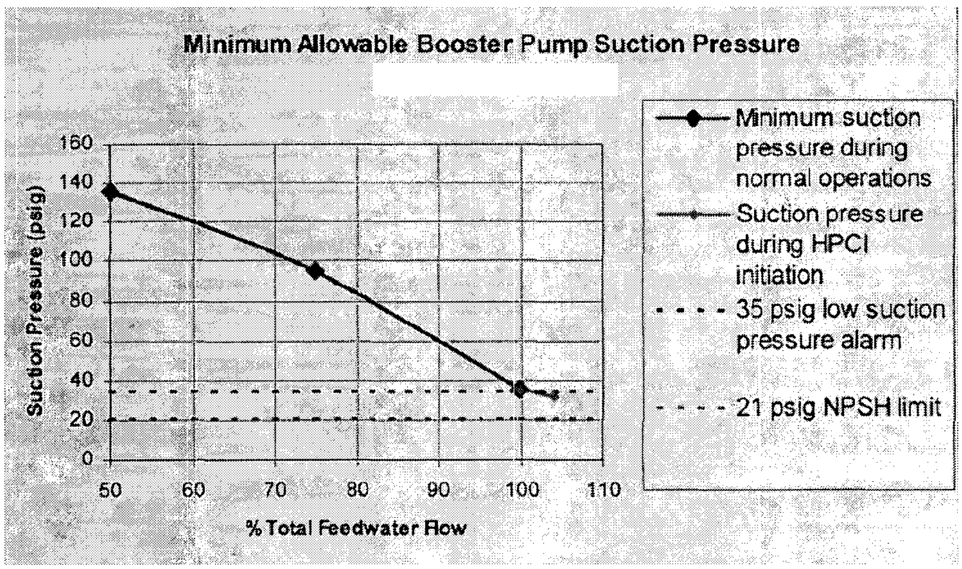
QUESTION 61

The plant is operating at 90% power with the following:

- Two Condensate pumps are in service
- Four (4) Condensate Prefilters are in service
- Five (5) Condensate Demineralizers are in service
- One (1) Condensate Demineralizer is out of service for maintenance

Then, one Condensate Prefilter is removed from service for backwash, resulting in the following conditions:

- Feedwater pump suction pressure is 198 psig and steady
- Feedwater Booster pump suction pressure is 57 psig and steady



Which one of the following describes the actions, if any, for these conditions?

A. No actions are required.

B. AO-201, Condensate Prefilter System Bypass, may be throttled open to assure HPCI operability.

C. AO-201, Condensate Prefilter System Bypass, may be throttled open to prevent Feedwater pump suction nozzle cavitation.

D. A third Condensate pump must be placed back in service until all six (6) Condensate Demineralizers are in service to assure HPCI operability.

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 B. Both the FW pump and FWBP suction pressures are below the required values for HPCI operability, requiring AO-201 to be throttled open. Plausible if the candidate mis-interprets the graph, or believes that since conditions are stable, and not degrading, continued operation is acceptable.

B. CORRECT B is correct – OP-15C attachment 4 requires AO-201 to be throttled to maintain FWBP suction pressure above the given curve. Additionally, OP-15A P&L 20 and OP-15C P&L 11 reference attachment 4. OP-15A P&L 5.0 requires FW pump suction pressure above 200 psig.

C. C is incorrect -- OP-15C attachment 4 requires AO-201 to be throttled open to maintain FWBP suction pressure above the given curve. This it to assure HPCI operability, not specifically prevent cavitation. Plausible if the candidate believes the suction pressure requirement is based on cavitation concerns.

D. D is incorrect -- Although restarting the Condensate pump would restore system parameters to acceptable values, it is not procedurally required. An allowed alternative is to open AO-201. Plausible if the candidate does not know that AO-201 can be opened under these conditions.

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

Learning Objective: N/A

Question source: NEW

Question History: NEW

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

10CFR 55 41.7

QUESTION 62

A plant shutdown is in progress with the following:

- Feedwater pump 11 is operating with the low flow control valve in automatic
- The Reactor Water Cleanup System is operating with two pumps in service
- Then, an Operator establishes RWCU reject flow to the Main Condenser

Which one of the following describes the effect of the RWCU reject flow on calculated core thermal power (CTP) and Non-Regenerative Heat Exchanger (NRHX) outlet temperature?

	Calculated CTP	NRHX Outlet Temperature
A.	Lowers	Lowers
B.	Lowers	Rises
C.	Rises	Lowers
D.	Rises	Rises

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. Raising RWCU reject flow rate raises the sensed RWCU flow rate to the CTP calculation. This causes calculated CTP to rise. Raising RWCU reject flow rate reduces the ability of the RHX to lower water temperature. This leads to higher NRHX outlet temperatures. Plausible if candidate does not fully understand these system effects.

B. B is incorrect. Raising RWCU reject flow rate raises the sensed RWCU flow rate to the CTP calculation. This causes calculated CTP to rise. Plausible if candidate does not fully understand the CTP calculation and its relation to RWCU.

C. C is incorrect. Raising RWCU reject flow rate reduces the ability of the RHX to lower water temperature. This leads to higher NRHX outlet temperatures. Plausible if candidate does not fully understand this system effect.

D. D is correct. Raising RWCU reject flow rate raises the sensed RWCU flow rate to the CTP calculation. This causes calculated CTP to rise. Raising RWCU reject flow rate reduces the ability of the RHX to lower water temperature. This leads to higher NRHX outlet temperatures.

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 CO_2 system for the Powerboard 102 room. Which one of the following describes how personnel exposure to CO_2 is prevented?

The horn and siren alarm for ___(1)___ before CO_2 discharges, and wintergreen oil is discharged ___(2)___ the CO_2.

	(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 Powerboard 102 room has a 30 second time delay with a predischarge alarm horn and light. 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 timer is set for 30 seconds. This is a plausible distracter for those candidates that are unsure of the timer design.

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

The plant is operating at 100% power when the following events occur:

- A seismic event triggers a large break LOCA that provides valid automatic initiation signals to the Containment Spray System
- Containment Spray Pumps 111 and 121 are rendered inoperable and are not able to be used

Which one of the following describes the ability of Containment Spray to provide rated spray flow in this situation?

The remaining ___(1)___ capacity Containment Spray pumps provide flow that is ___(2)___.

	(1)	(2)
A.	50%	Less than design basis spray flow and can not meet the requirements for the Appendix J Containment Spray Water Seal.
B.	100%	Less than design basis spray flow and can not meet the requirements for the Appendix J Containment Spray Water Seal.
C.	50%	Greater than or equal to design basis spray flow and can meet the requirements for the Appendix J Containment Spray Water Seal.
D.	100%	Greater than or equal to design basis spray flow and can meet the requirements for 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 – 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.

B. B is incorrect – The failure of pumps 111 and 121 will result in pumps 112 and 122 feeding 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.

C. C 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.

D. CORRECT D is correct –Each containment spray pump is rated for 100% flow. Either of the two operable 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.

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

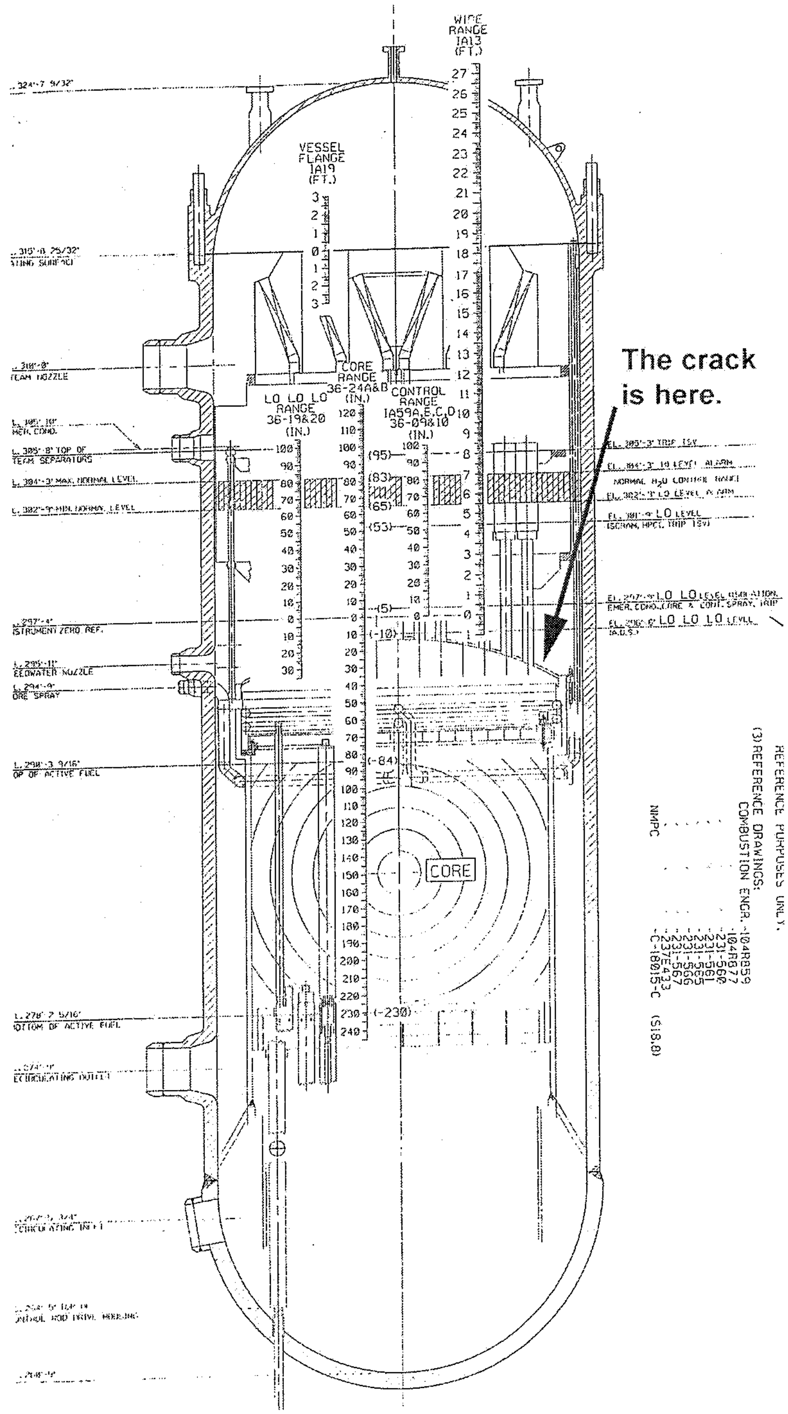
10CFR 55 41.7

QUESTION 65

The plant is operating at 100% power when a large crack in the shroud head occurs as indicated in the picture on the next page.

Per N1-SOP-1.5, Unplanned Reactor Power Change, which one of the following describes the changes in the following parameters as a result of this failure?

	Reactor Power	Recirc Suction Temp.
A.	Decrease (↓)	Increase (↑)
B.	Increase (↑)	Decrease (↓)
C.	Decrease (↓)	Decrease (↓)
D.	Increase (↑)	Increase (↑)



The crack is here.
WIDE RANGE IA13 (FT.)
VESSEL FLANGE IA19 (FT.)
CORE RANGE 36-24A&B (IN.)
LO LO LO RANGE 36-19&20 (IN.)
CONTROL RANGE IA59A,B,C,D 36-09&10 (IN.)
CORE
REFERENCE PURPOSES ONLY.
(3) REFERENCE DRAWINGS:
COMBUSTION ENGR.
NMPC

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 shroud 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).

B. B is incorrect – This is a plausible distracter for candidates who believe the crack in the shroud 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 shroud 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 shroud 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 CRO on day shift on May 10

Which one of the following describes the Control Room Logs that must be reviewed before assuming the shift, in accordance with CNG-OP-1.01-2002?

A. All Control Room Logs back to and including day shift on May 3. Further back is not required and anything less is not appropriate.

B. All Control Room Logs back to and including day shift on May 5. Further back is not required and anything less is not appropriate.

C. All Control Room Logs back to and including day shift on May 8. Further back is not required and anything less is not appropriate.

D. Just the day and night shift Control Room logs for May 9. 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. 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.

B. Incorrect because the last 5 days is not the requirement, but plausible if the candidate does not know the requirement then the last 5 days seems reasonable.

C. Correct – Per CNG-OP-1.01-2002, the operator is required to "Review the Control Room logs going back to their last watch, or 48 hours, whichever is shorter."

D. 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.

References: CNG-OP-1.01-2002 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.

Which one of the following describes the limitation on isolating recirculation loops and the reason for that limitation?

The suction and discharge valves of at least __(X)__ recirculation loop(s) shall be full open unless the reactor vessel is flooded to above the main steam line nozzles, or the steam dryer and separator are removed. This is to __(Y)__.

	(X)	(Y)
A.	One	Ensure reactor water level instrument readings indicate actual level in the core region.
B.	One	Prevent thermal stratification while shutdown with fuel in the Reactor Vessel.
C.	Two	Ensure reactor water level instrument readings indicate actual level in the core region.
D.	Two	Prevent thermal stratification while shutdown with fuel in the Reactor Vessel.

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. 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.

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. 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.

D. 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.

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

Given the following:

- 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 as follows:

Day	Date	Work Hours
Monday	10/1	1800-0600
Tuesday	10/2	1800-0600
Wednesday	10/3	1800-0600
Thursday	10/4	Off
Friday	10/5	Off
Saturday	10/6	0600-1800
Sunday	10/7	0600-1800
Monday	10/8	0600-1800

- The plant is conducting a startup after a plant trip on the previous Friday.
- You stay a total of three (3) hours past end of shift on Monday, 10/1, due to an extensive turnover.

Which one of the following will satisfy the working hour requirements for you to work your scheduled shifts?

A. No action required, because turnover time is not included in calculating working hours.

B. Notify your supervisor that your work hour limits will be exceeded and obtain verbal approval from your supervisor to work your scheduled hours.

C. Notify your supervisor that your work hour limits will be exceeded. You have up to your shift on Tuesday, 10/2, to have an overtime deviation request approved by the Plant General Manager.

D. Notify your supervisor that your work hour limits will be exceeded. You have up to your shift on Monday, 10/8, to have an overtime deviation request approved by the Plant General Manager.

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 turnover time of greater than 1 hour is counted towards calculating work hours.

B. Incorrect but plausible because this is the correct response for personnel who do NOT perform or directly supervise those who perform safety related functions.

C. 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.

D. 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.

References: GAP-FFD-02 / CNG-SE-1.01-XXXX 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 identifies all of the personnel the Operating Permit Section Holder is REQUIRED to contact prior to performing manipulation(s) of tagged components under an Operating Permit Tag?

A. All Work Order Holders

B. All Work Order Holders and Operating Permit Tagout Holders

C. All Worker Tracking List Holders and the Tagging Authority

D. All Worker Tracking List Holders, the Tagging Authority and all Work Order 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. 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

C. Incorrect because the Section Holder needs permission of the work order holders, worker tracking list holders, and the Tagging Authority.

D. 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. 5.4.N requires permission from the Tagging Authority.

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

The plant is operating at 100% power when an unexpected transient occurs.

Which one of the following describes a resulting condition that will violate a 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 1.21

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 1345 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 operating at 100% power with the following:

- Annunciator H1-1-7, OFF GAS HIGH RADIATION, alarms
- The reactor operator reports that both Offgas Radiation Monitors are at the Hi 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. Verify Mechanical Vacuum Pump trip and isolation

D. Reduce Reactor power as necessary to control radiation levels

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. Incorrect because the step to verify mechanical vacuum 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.

D. Correct – With an off gas high rad alarm in, sop-25.2 directs to reduce power to control radiation levels.

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 evacuated due to a fire, with the following:

- Operators are required to enter a radiologically posted area in order to manually close Primary Containment Isolation Valves.
- The highest dose rate in the area is 1500 mR/hr.

Which one of the following describes the radiological requirements for the area and the entry?

This area is required to be posted as a ___(1)___ and personnel entering the area are required to maintain each entrance closed and locked except ___(2)___.

	(1)	(2)
A.	High Radiation Area	when inside the area so personnel are not prevented from leaving the area.
B.	Locked High Radiation Area	when inside the area so personnel are not prevented from leaving the area.
C.	Locked High Radiation Area	for periods of ingress or egress, unless guarded to prevent unauthorized entry.
D.	Very High Radiation Area	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 resulted in the following:

- Reactor water level is 63 inches and rising slowly
- Reactor pressure is 1090 psig and steady
- Reactor power is 50% and steady
- Drywell pressure is 3 psig and rising slowly
- Drywell average temperature is 145°F and rising slowly
- Torus water level is 11.30 feet and rising slowly
- Torus temperature is 82°F and rising slowly
- Annunciator H2-2-2, R BUILDING EQUIP DRAIN LEVEL HIGH, is in alarm
- An operator reports Reactor Building Equipment Drain Tank water level is one (1) foot below floor level.

Which one of the following lists the appropriate Emergency Operating Procedure(s) to be entered?

A. EOP-2, EOP-3, EOP-4, EOP-5

B. EOP-2, EOP-3, EOP-4, only

C. EOP-2, EOP-4, only

D. EOP-4, only

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. CORRECT B 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.

C. C 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.

D. D 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.

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 100% power with the following:

- The Shift Manager has declared an UNUSUAL EVENT due to a 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 damage control team.

B. Immediately report to the OSC and wait to be assigned to a damage control 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: X
Comprehensive/Analysis:

10CFR 41.10

QUESTION 75

The plant is operating at 100% power when the following alarm annunciates:

- F2-1-2, REACT RECIRC PUMP-MOTOR 12

Recirc Pump 12 seal indications are as follows:

- High Pressure Seal pressure is 380 psig
- Low Pressure Seal pressure is 350 psig

Which one of the following describes the status of Recirculation Pump 12?

A. The breakdown orifice between the seals has plugged

B. Only the High Pressure Seal has failed

C. Only the Low Pressure Seal has failed

D. Both Seals 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 pressure would remain high on HP seal however LP seal would have a lower pressure if the breakdown orifice is plugged.

B. Incorrect but plausible since the pressures would be about equal in both HP seal and LP seal if HP seal had failed. However the pressure would have been much higher (approx 1000psig) if HP seal had failed.

C. Incorrect but plausible since the pressure would have remained 1000 psig on HP seal and LP seal would have been lower than 500 psig if only LP seal failed. if the

D. Correct – Normal Pressure on the HP and LP 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 plant was scrammed due to rising Drywell pressure.

One (1) minute after the scram, the following conditions exist:

- RPV water level indicates 0 inches on the Narrow Range GEMACs
- RPV water level indicates 0 inches on the Hi-Lo/Lo-Lo Rosemounts (Yarways)
- RPV water level indicates 1.0 feet on Wide Range
- RPV water level indicates upscale on both Lo-Lo-Lo Rosemounts
- Both Fuel Zone indicators are indicating +110 inches
- Reactor pressure is 350 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 injecting
- Recirc Pump 15 has failed to trip

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 (zero).

Wide range level indication (36-94) is not usable when indicating less than 1.5 feet at the current drywell temperature. With the current 1.0 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 because Recirc Pump 15 is still running.

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. None of the given conditions require entry into EOP-8 yet.

C. C is incorrect – Current plant conditions do not meet the criteria that **require** initiation of containment sprays.

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. EOP-7 requires flooding to the main steam lines.

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 resulted in the following Containment parameters over the past four minutes:

	08:01	08:02	08:03	08:04
Drywell Pressure (psig)	6.0	9.0	11.0	14.0
Torus Pressure (psig)	5.0	8.0	10.0	13.0
Drywell Temperature (°F)	225	250	276	302

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 Core Shuffle Phase II in progress, when the following events occur:

- Shutdown Cooling (SDC) is lost
- Actual temperature measurements indicate Reactor water temperature is projected to exceed 140°F

Which one of the following is required, in accordance with N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal?

A. Perform a time to boil estimation.

B. Verify Secondary Containment established.

C. Secure all Operations with the Potential for Draining the Reactor Vessel (OPDRVs).

D. Return any core component being transferred to the nearest storage location in Spent Fuel Pool OR Core.

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 also covers actions for loss of vessel inventory while refueling.

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 de-energized due to a fault
- All available systems are injecting
- RPV water level is -70 inches and steady
- RPV pressure is 200 psig and lowering slowly
- Torus pressure is 17 psig and lowering slowly
- Torus temperature is 130°F and rising slowly
- Torus level is 12 feet and steady
- Drywell pressure is 21 psig and lowering slowly
- Drywell temperature is 250°F and lowering slowly
- Containment Spray Pump 112 has just tripped on over-current

Which one of the following is required?

A. Blowdown per N1-EOP-8

B. Vent the Torus per N1-EOP-4.1, Section 2

C. Cool the Torus per N1-EOP-1, Attachment 16

D. Spray the Containment with Raw Water per N1-EOP-1, Attachment 17

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. 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.

B. 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.

C. 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.

D. CORRECT - Per EOP-4, Step PCP-4 can use external source (raw water) to spray as long as you stay below PCPL.

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 43.5

QUESTION 80

Which one of the following sets of conditions presents a situation where it is expected that an RPV Blowdown will cause one of the following to be exceeded:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the containment can be opened and closed?

A. RPV pressure is 950 psig
Torus temperature is 120°F
Torus water level is 11.5 feet

B. RPV pressure is 700 psig
Torus temperature is 130°F
Torus water level is 11 feet

C. Drywell temperature is 350°F
Drywell pressure is 3 psig

D. Torus pressure is 30 psig
Primary Containment water level is 30 feet

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.

B. B is incorrect – HCTL is not violated in answer B. Additional depressurization with ERV's will maintain the plant below the HCTL limit. 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. No indication is given that HCTL is being violated.

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. No indication is given that HCTL is being violated.

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 N1-SOP-21.2, Control Room Evacuation, is entered.

The following events occur as a result 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 ERVs 111 and 112 to spuriously open until their fuses were pulled
- Reactor water level dropped due to the ERV actuation until an automatic Vessel and Containment Isolation occurred
- Reactor water 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

Which one of the following describes the most restrictive time 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 to the State and County 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-21.2 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 41.10

QUESTION 82

The plant is operating at 100% power when the Fire Detection System senses a fire in Hazard C-2113, Powerboard 103 Room.

The following Alarm Detection Zones are received at Main Fire Panel 2:

- DX-2113A
- DX-2113B

An operator in the area reports the following:

- The local horn and light have actuated
- CO_2 has discharged into the area

Which one of the following is required at this time?

A. Verify Control Room is at a positive pressure relative to outside air pressure to prevent CO_2 leaking into the Control Room.

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 1, 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 – Plant conditions warrant declaration of an Alert per EAL 8.3.5 due to CO2 discharge. Plausible if candidate believes a UE is warranted based on the fire. UE declaration is further incorrect because no indication is present that the fire has been going for greater than 15 minutes.

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

QUESTION 83

The plant is operating at 80% power when a seismic event occurs. A Torus water leak results in the following:

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, is in alarm
- Computer point F188 NE RB CORNER RM WTR LVL HIGH is in alarm
- An operator reports water level in the Northeast Corner Room is 5 feet and slowly rising

Which one of the following describes 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, EOP Bases   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 operating at 100% power with fuel leaks identified. The following sequence of events occurs:

Time	Event
1500	Annunciator 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 mrem/hr
1506	Manual reactor scram inserted
1507	EMERG COND RMON 111 and 112 are at 15 mrem/hr and lowering slowly

Which one of the following is the next required action?

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 an RPV Blowdown per EOP-8 as directed by EOP-5.

D. Perform an 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

The plant has experienced a transient with the following conditions:

- A steam leak has occurred from Emergency Condenser 11 into Secondary Containment.
- Emergency Condenser 11 is in service for pressure control.
- N1-EOP-5, Secondary Containment Control, has been entered.
- Step SC-5 states:

Isolate all discharges into affected areas <u>except</u> systems needed for:

- Damage control
- Other EOP actions

SC-5

Which one of the following describes conditions in which Emergency Condenser 11 should be left in service?

A.
- The MSIVs inadvertently isolated during the transient.
- CRD has tripped and is not able to be put back in service.
- Emergency Condenser 12 is tagged out for maintenance.

B.
- An ATWS is in progress with the MSIVs failed closed.
- Torus level is 6 feet and lowering.
- Emergency Condenser 12 is also in service with Reactor pressure 950 psig and slowly rising.

C.
- Reactor water level is 75 inches and lowering slowly.
- The MSIVs inadvertently isolated during the transient.
- No sources of high pressure feed are available.

D.
- An ATWS is in progress with the MSIVs failed closed.
- Torus level is 11 feet and steady.
- Emergency Condenser 12 is also in service with Reactor pressure 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 MSIVs isolated and EC 12 out of service pressure control can be transitioned to ERVs 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 and the ERVs are available.

B. CORRECT B is correct – Both ECs are in service and pressure is slowly rising under this condition. No other pressure control mechanisms are available for these conditions with the MSIVs closed and torus level being too low to utilize ERVs. Additionally, current Torus level requires a blowdown per EOP-8 which requires both EC's to be in service. Therefore EC 11 must remain in service to depressurize the reactor.

C. C is incorrect. With the reactor isolated, the condenser unavailable for pressure control and no high pressure feed available for level control, use of ECs is desirable to control pressure and minimize inventory loss. The leak on the EC Loop 11 can be isolated and EC Loop 12 placed in service to maintain plant conditions. Plausible if candidate does not realize has EC 12 available.

D. D is incorrect – ERVs are available for pressure control and therefore EC 11 can be removed from service and isolated. Plausible because with pressure rising the candidate may think EC #11 should not be removed from service. 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 describes the bases for the minimum required fuel supply for EDG operation?

The minimum required fuel supply for EDG operation is based on....

A. Both EDGs operating following a LOCA for two days.

B. A single EDG operating following a LOCA for two days.

C. Both EDGs operating following a LOCA for four days.

D. A single EDG operating 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, EDGs, 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, EDGs, 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 at 100% power with the following:

- Maintenance was performed on Emergency Condenser elbow flow meters 36-06C and 36-06D.
- On the next shift, valves 36-116 and 36-60 were both found stuck in the closed position.

Which one of the following describes the action required by Technical Specifications that must be implemented first?

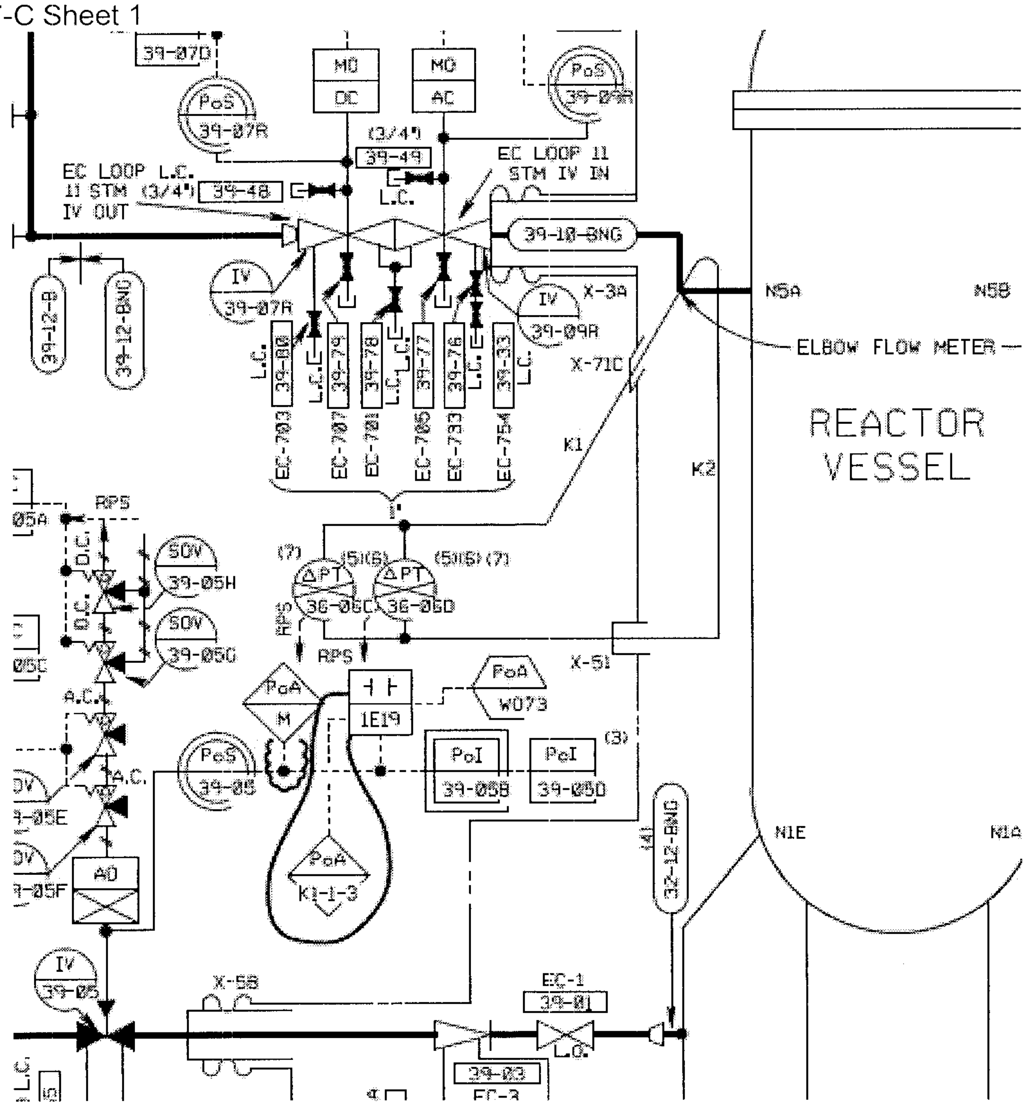
A. Restore operability of the affected system within 7 days, and perform the additional required surveillances.

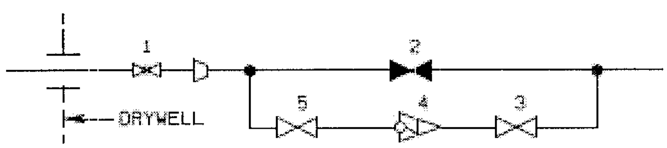
B. Place the inoperable channel(s) in the tripped condition within 24 hours

C. Place the inoperable channel in one trip system in the tripped condition within one hour.

D. Commence a normal shutdown within one hour, and be in cold shutdown within 10 hours.

C-18017-C Sheet 1





OP. NO.	PEN	1	2	3	4	5	REF.
EC-1001	X-71C K1	36-116	36-59	36-58	36-57	36-56	△PT 36-06C △PT 36-06D
EC-1003	X-51 K2	36-60	36-64	36-63	36-62	36-61	△PT 36-06C △PT 36-06D
EC-1002	X-54 K1	36-65	36-69	36-68	36-67	36-66	△PT 36-06A △PT 36-06B
EC-1004	X-72C K2	36-70	36-74	36-73	36-72	36-71	△PT 36-06A △PT 36-06B

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 because the insertion of a trip signal is required first. Plausible if the candidate goes directly to 3.1.3 and assumes 36-06A being inoperable makes the EC inoperable, without addressing TS 3.6.2. This is the action for one EC inoperable.

B. Incorrect because the insertion of a trip signal in one channel is required in one hour. 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.

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 instruments being inop does not inop both ECs, Plausible if the candidate goes directly to 3.1.3 and assumes 36-06C and 36-06D being inoperable makes both ECs inoperable. This is the action for both ECs inoperable.

References: TS 3.1.3, 3.6.2, PID C-18017-C Student Ref: TS 3.1.3, 3.6.2

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 with the following conditions:

- All IRM range switches are on Range 10
- The Mode Switch is in STARTUP

Then, an event occurs which results in reduced recirculation pump flow:

- Total Core Flow is 19 Mlbm/hr
- All IRMs remain on Range 10

Which one of the following states the Technical Specification requirements due to this condition and the basis for this requirement?

	Tech Spec Requirements	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

The plant is shutdown with in-vessel maintenance in progress. The following conditions exist:

- SRMs 11, 12 and 14 are fully inserted and operable
- SRM 13 is inoperable
- The Reactor Mode Switch is in REFUEL
- Core cell 14-31 is adjacent to SRM 12
- Core cell 14-31 is defueled in preparation for removal of Control Rod Blade (CRB) 14-31 to the spent fuel pool
- CRB 14-31 is fully withdrawn and uncoupled

Before CRB 14-31 is removed, the following events occur:

- SRM 14 is withdrawn during testing and counts drop to 90 cps
- SRM 12 counts have dropped to 50 cps following defueling of cell 14-31

Note: Core location map provided on following page.

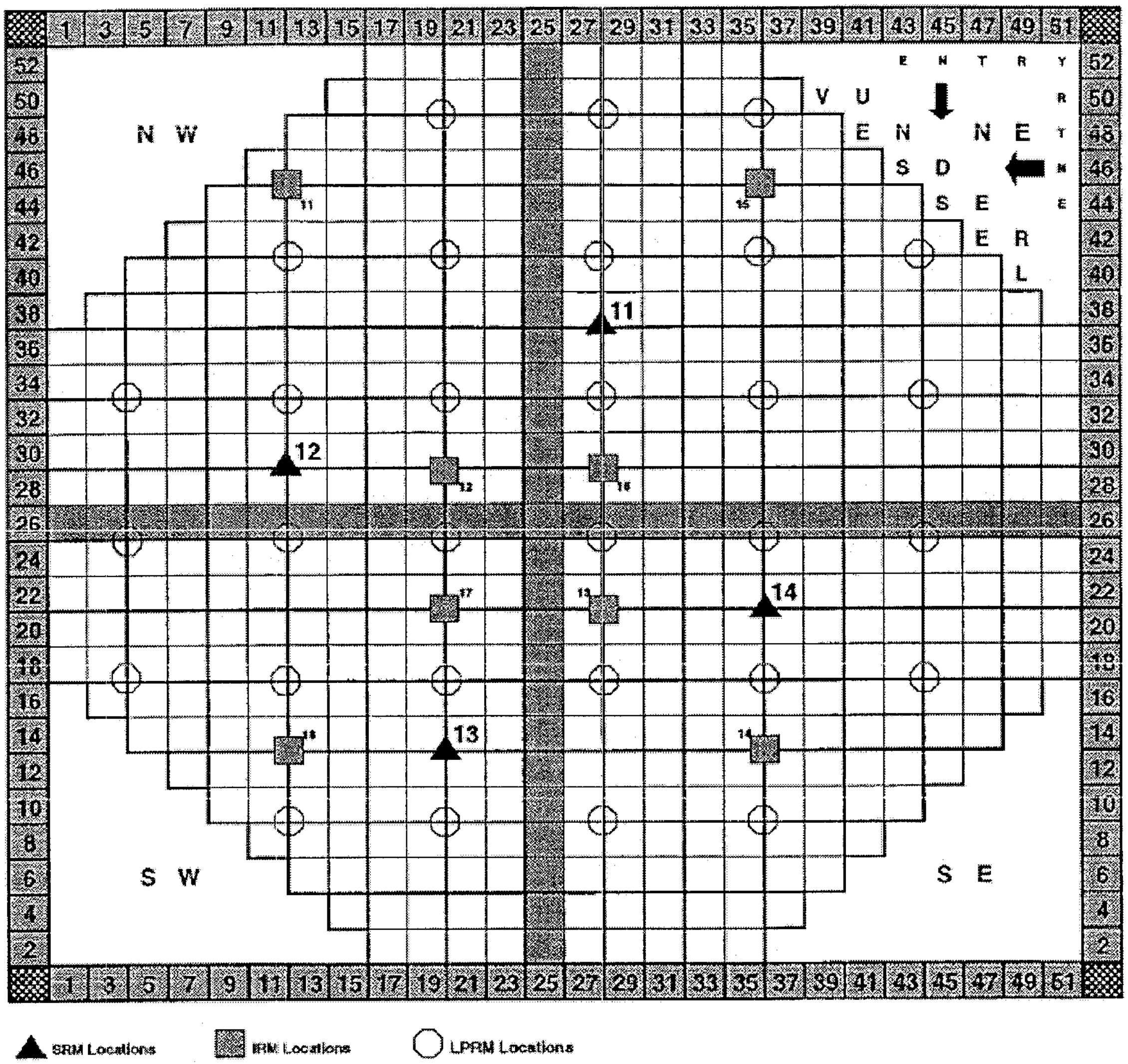
Which one of the following statements is correct in response to these conditions?

A. The control rod blade may be moved into the fuel pool with the current conditions.

B. Only SRM 12 must be declared inoperable and CRB 14-31 may be placed in the fuel pool only after SRM 12 is made operable.

C. Only SRM 14 must be declared inoperable and CRB 14-31 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 14-31 may be placed in the fuel pool only after SRM 12 AND SRM 14 are made operable.


1 3 5 7 9 11 13 15 17 19 21 23 25 27 29 31 33 35 37 39 41 43 45 47 49 51
52 50 48 46 44 42 40 38 36 34 32 30 28 26 24 22 20 18 16 14 12 10 8 6 4 2
N W
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11
12
13
14
SRM Locations
IRM Locations
LPRM Locations

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 41.10

QUESTION 90

The plant is operating at 95% power with the following:

- Four Recirc loops are in service
- Recirc pump 14 is shutdown and isolated for maintenance

Then, Recirc pump 11 trips due to an electrical fault.

Which one of the following describes the required action(s)?

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 the MCPR safety limit.

C. Within 1 hour, close Recirc pump 11 discharge or suction valve, or then take action to insert control rods.

D. Within 12 hours, close Recirc pump 11 discharge or suction valve, or then take action to insert control rods.

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. Tech Spec 3.1.7.e requires thermal limit adjustments for 3 loop operation. 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 eight 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 eight 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.

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) may believe that 12 hours are available 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

QUESTION 91

The plant is operating at 100% power when the following alarm is received:

- H1-1-7, OFF GAS HIGH RADIATION

Chemistry has confirmed the alarm. Offgas radiation monitor alarm setpoints and actual reading are as follows:

- Alarm setpoint (Hi) - 150 mR/hr
- Alarm setpoint (Hi-Hi) - 1500 mR/hr
- Actual readings - 6600 mR/hr and slowly rising

Two minutes later, the following alarms are received:

- F1-2-7, MAIN STEAM RAD MONITOR CH 11 HI/LO
- F4-2-2, MAIN STEAM RAD MONITOR CH 12 HI/LO

Main Steam Line radiation monitor alarm setpoints and actual readings are as follows:

- Alarm setpoints - 900 mR/hr
- Actual readings - 1500 mR/hr and slowly rising

Which one of the following describes the required operator action(s) for the high radiation conditions?

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 Hi-Hi 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 is less than 3.75 time normal full power background, and therefore do not require scram and vessel isolation.

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

QUESTION 92

The plant is shutdown for a refueling outage with the following:

- Phase I core shuffle is in progress
- Multiple Operations with the Potential to Drain the Reactor Vessel (OPDRVs) are in progress
- Then, the Control Room Emergency Ventilation System (CREVS) is declared inoperable

Which one of the following describes (1) the required Tech Spec Action and (2) an operation that can be excluded as an OPDRV without further approvals?

	(1)	(2)
A.	Suspend all OPDRVs if CREVS is not returned to operable status within 7 days	CRD Mechanism removal and replacement per the approved vendor procedure
B.	Suspend all OPDRVs if CREVS is not returned to operable status within 7 days	Recirc Pump #14 discharge valve replacement which requires an air bladder installed on the discharge piping
C.	Immediately suspend all OPDRVs until CREVS is restored to operable status	CRD Mechanism removal and replacement per the approved vendor procedure
D.	Immediately suspend all OPDRVs until CREVS is restored to operable status	Recirc Pump #14 discharge valve replacement which requires an air bladder installed on the discharge piping

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.

Additionally, TS 3.4.5.h requires suspending OPDRVs if TS 3.4.5.e is not met. TS 3.4.5.e is met until CRATS is inop for 7 days.

B. B 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.

C. C is incorrect – CRD removals and replacements are a covered exception to the definition of an OPDRV, this activity may continue with the CREVS system inoperable However, OPDRVs are not required to be suspended immediately. Tech Spec 3.4.5e allows seven (7) days to restore the system to an operable status before entry into TS action 3.4.5h. Entry into 3.4.5h requires immediately initiating action to suspend OPDRVs.

D. D 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. Additionally, Tech Spec 3.4.5e allows seven (7) days to restore the system to an operable status before entry into TS action 3.4.5h. Entry into 3.4.5h requires immediately initiating action to suspend OPDRVs

References: N1-OP-34, C-18006-C, C-18009-C, TS 3.4.5 Student Ref: TS 3.4.5

Learning Objective: N/A

Question source: New

Question History: New

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

10CFR 55	41.5	45.6

QUESTION 93

Phase II core shuffle is in progress with all control rods fully inserted. The following two moves have been made.

- 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.
- 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 core.

During both moves, the following indications were observed during the entire time of the moves:

- In the Control Room, the Rod Block Monitor Panel REFUEL INTERLOCK light is NOT illuminated.
- On the Refuel Bridge, the ROD BLOCK INTERLOCK light is NOT illuminated.

Which one of the following describes the refueling interlock/limit switch that has failed and when the rod block should have been in effect?

	Interlock/Limit Switch Failure	Rod Block in Effect
A.	The refuel bridge reverse motion stop interlock	When the fuel bundle was over the core.
B.	The refuel bridge reverse motion stop interlock	When the DBG was over the core.
C.	The "over core" limit switch failed on the refuel bridge.	When the fuel bundle was over 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 reverse motion stop 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 reverse motion stop 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

QUESTION 94

The plant is starting up after an outage. The following chemistry samples have been taken over the past three days. Results were available at 0800 each day, but are just now being analyzed for Technical Specification implication.

	Day 1	Day 2	Day 3
Reactor coolant pH	8.9	9.1	9.1
Reactor coolant conductivity (μmho/cm)	1.19	0.57	0.19
Reactor coolant chlorides (ppb)	175	110	25
Reactor coolant sulfates (ppb)	25	20	18
Steam flow (lbm/hr)	50,000	2,000,000	4,500,000

Which one of the following describes the Technical Specification implications?

A. No Technical Specifications have been exceeded. No actions are required.

B. A shutdown is required. The shutdown should have been initiated at 0900 on Day 1.

C. A shutdown is required. The shutdown should have been initiated at 0900 on Day 2.

D. A shutdown is required. The shutdown should have been initiated at 0900 on Day 3.

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 sees 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

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 from a refuel outage is in progress, with the following:

- Reactor power is 75%
- Feedwater pump 13 is the only Feedwater pump in service
- Core flow is 60%
- Five Recirc pumps are in service
- Recirc MG set scoop tube limits are set at 102.5%
- Tau (τ) = 0.8
- MCPR is determined to be 1.50

Which one of the following describes when the first Technical Specification action is required, if any?

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 2a) 1.497 X (Kf from figure 2e) 1.07=1.602. 1.602>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 Without safety limits.

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

An Immediate Change was generated for a Special Operating Procedure (SOP). The Plant Management Staff approval has been completed.

Which one of the following identifies:

(1) the individual or group required to approve the Immediate Change <u>BEFORE</u> it can be implemented, and
(2) the individual or group required to provide the <u>FINAL</u> approval within 14 days,

in accordance with CNG-PR-1.01-1011, Control of Station-Specific Procedure Change Process?

	(1)	(2)
A.	Any active SRO	Approval Authority
B.	Any active SRO	PORC
C.	PDU / Procedure Sponsor	Approval Authority
D.	PDU / Procedure Sponsor	PORC

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. Correct – CNG-PR-1.01-1011 requires approval by an active SRO prior to implementation, and then approval from the "Approval Authority" within 14 days. The Approval Authority can be the Ops Manager, GSO, Responsible Procedure Owner or Alternate Responsible Procedure Owner.

B. Incorrect because the final review within 14 days is required to be completed by the "Approval Authority", not PORC.

C. Incorrect because any active SRO can approve the initial implementation of the Immediate Change. This answer is plausible because the "Approval Authority is required to approve this procedure change within 14 days.

D. Incorrect because the final review within 14 days is required to be completed by the "Approval Authority", not PORC, and the initial review must be completed by an active SRO.

References: CNG-PR-1.01-1011 Student Ref: None

Learning Objective: N/A

Question source: NMP1 2004 NRC

Question History: None

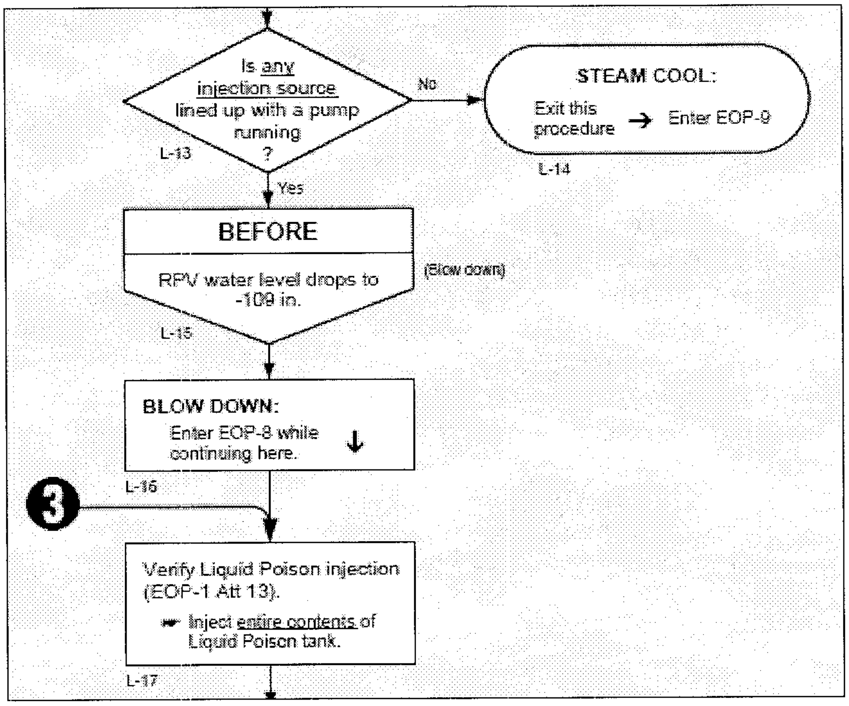
Cognitive level: Memory/Fundamental knowledge: X
Comprehensive/Analysis:

10CFR 43.3

QUESTION 98

A loss of coolant accident has occurred.

N1-EOP-2, RPV Control, has been entered with Reactor level continuing to drop.



Which one of the following describes 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 loss of 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: EOP-2, 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: X
Comprehensive/Analysis:

10CFR 55 41.5 45.6

QUESTION 99

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

Time (hh:mm)	Event
00:10	Plant conditions justify the declaration of an ALERT
00:17	Plant conditions change and only meet the UNUSUAL EVENT threshold. Emergency declaration has not yet been made.

Which one of the following describes the latest time at which the emergency declaration must be made and the level of emergency that must be declared and reported, in accordance with EPIP-EPP-20, Emergency Notifications?

	Latest Time For Declaration	Level of Emergency Declaration and Report
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 the UNUSUAL EVENT. The transmittal form should indicate the ALERT conditions that occurred.
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

QUESTION 100

Step TL-4 of N1-EOP-4, Primary Containment Control, directs to maintain Torus water level below 13.5 feet.

Which one of the following describes the basis for maintaining Torus level below 13.5 feet?

A. Maintaining Torus level below 13.5 feet ensures that the Torus has adequate capability to be vented.

B. The Torus boundary design load would be exceeded if ERVs were opened with the Torus level above 13.5 feet.

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 feature of the Primary Containment is assumed to function as designed only when Torus 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 vent is located at 27 feet, and is not the basis for maintaining below 13.5 feet. Plausible if the candidate does not know the location of the Torus vent connection.

B. B 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.

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