Question 53
Part 1 - Original Question

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA2.16
	Importance Rating	3.0

Plant Fire On-site

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Vital equipment and control systems to be maintained and operated during a fire

Proposed Question: #53

The plant has experienced a significant fire with the following:

- The Reactor has been scrammed.
- A Reactor cooldown has been performed.
- Reactor pressure is 50 psig and stable.
- Shutdown Cooling is unavailable due to the fire.
- Continued Reactor cooldown is desired.

Which one of the following describes the Alternate Shutdown Cooling lineup to be used to continue the Reactor cooldown, in accordance with N1-SOP-21.1, Fire in Plant?

A. Initiate both Emergency Condensers while injecting to maintain Reactor water level 53 to 95 inches.

B. Raise Reactor water level to the Main Steam Lines to circulate water to the Torus through ERVs.

C. Maximize Reactor Water Cleanup reject flow while injecting to maintain Reactor water level 53 to 95 inches.

D. Raise Reactor water level to the Main Steam Lines to circulate water to the Main Condenser through Turbine Bypass Valves.

Proposed Answer: B

Explanation: N1-SOP-21.1 contains specific guidance on how to establish an Alternate Shutdown Cooling if normal Shutdown Cooling is unavailable and Reactor pressure is too low to further cooldown by steaming through ERVs (approximately 50 psig). This Alternate Shutdown Cooling lineup isolates the Main Steam lines and Emergency Condenser steam lines, then raises Reactor water level to the Main Steam lines to inject through the ERVs into the Torus. Torus Cooling is placed in service to remove decay heat.

A. Plausible – With 50 psig of steam pressure left in the Reactor, initiating Emergency Condensers would provide some decay heat removal, however this is not the Alternate Shutdown Cooling lineup called for by N1-SOP-21.1.
C. Plausible – RWCU reject with injection to the Reactor does provide a flow path that would remove some decay heat, however this is not the Alternate Shutdown Cooling lineup called for by N1-SOP-21.1.
D. Plausible – The Alternate Shutdown Cooling method in N1-SOP-21.1 does required raising Reactor water level to the Main Steam lines, but to circulate water to the Torus, not the Main Condenser.

Technical Reference(s): N1-SOP-21.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP211C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Question 53
Part 2 - Justification

Analysis:

The stem conditions present a condition where SDC is unavailable due to a fire and is asking what the decay heat removal strategy is for this condition, in accordance with N1-SOP-21.1, Fire in Plant.

The question was developed and validated based on an old revision of N1-SOP-21.1, Fire in Plant. N1-SOP-21.1 was revised in October 2014 and the applicants were trained on the newer revision since that time. The new revision eliminated the keyed answer as an option.

N1-SOP-21.1 says that IF the fire endangers safe shutdown capability, THEN Scram the reactor (completed per the stem), Use N1-EOP-2, RPV Control as supplemental guidance, and maintain the reactor in stable hot shutdown condition until direction is given to transition to cold shutdown. The stem conditions indicate that direction has been given to transition to cold shutdown. This is performed using N1-EOP-2.

In N1-EOP-2, there are two strategies that are designed to be executed in conjunction with one another to ensure adequate decay heat removal.

The LEVEL Control strategy directs "Restore and Maintain RPV water level between 53 in. and 95 in. using one or more Preferred Injection Systems". (Step L-7)

The PRESSURE Control Strategy directs WAIT until the Shutdown Cooling Pressure interlock clears (120 psig). (Step P-4) It then states, IF Shutdown Cooling in unavailable (which is the case in this question), THEN Maintain RPV Pressure below 120 psig using one or more Alternate Pressure Control Systems. (Step P-5) The listed Alternate Pressure Control Systems include using ECs and RWCU Reject mode. (Step P-2) The stem states RPV pressure is 50 psig, therefore those actions apply.

Conclusions:

The decay heat removal strategy in N1-SOP-21.1 is to execute N1-EOP-2 as explained above. Both A and C are correct answers. Either would be equivalently correct.

Recommendation:

Recommend accepting both choices A and C as correct.

NRC Response

Comment accepted. Procedure N1-SOP-21.1 directs the operators to concurrently execute EOP-2, RPV Control. EOP-2 uses two strategies, performed in conjunction, to remove decay heat. The injection systems used for level control, and the alternate pressure control systems used for pressure control are listed in EOP-2, and match the descriptions of level and pressure control systems cited in choices A and C. Furthermore, actions described in B and D have no procedure basis, due to the N1-SOP-21.1 revision. In summary, A and C are both correct.

Question 69
Part 1 - Original Question

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.12
	Importance Rating	3.7

Knowledge of surveillance procedures.

Proposed Question: #69

N1-ST-Q1C, Core Spray (CS) 112 Pump and Valve Operability Test, is to be performed.

Which one of the following is allowed without causing "pre-conditioning" in accordance with GAP-SAT-01, Surveillance Test Program, and CNG-MN-4.01-1008, Pre/Post-Maintenance Testing?

A. Manually stroking isolation valves to verify they are properly seated.

B. Filling and venting the system prior to starting Core Spray pump 112.

C. Adjusting the packing on isolation valves to ensure no stem binding occurs.

D. Starting and running Core Spray pump 112 to bring it up to operating temperature.

Proposed Answer: B

Explanation: Per Attachment 2 of CNG-MN-4.01-1008, Pre/Post Maintenance Testing Requirements, filling and venting a system provided the venting operation has proper controls does not constitute preconditioning. Proper controls are provided directly in N1-ST-Q1C and similar surveillance tests.

A. Plausible – Isolation valve testing is part of this ST, however exercising the valves prior to the ST would constitute pre-conditioning.
C. Plausible – Isolation valve testing is part of this ST, however adjusting packing prior to the ST may affect stroke times and would constitute pre-conditioning.
D. Plausible – Core Spray pump 112 is operated as part of the ST, however it is required to be at normal standby temperature to avoid pre-conditioning.

Technical Reference(s): GAP-SAT-01, CNG-MN-4.01-1008 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-MN-4.01-1008-CT-01

Question Source: Bank – NMP2 2013 Audit #69

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Question 69
Part 2 - Justification

Analysis:

The stem conditions present a specific surveillance test (N1-ST-Q1C) and ask which of the choices could be done without causing "preconditioning".

Choice B is the keyed correct response and remains a correct response based on specific examples listed in CNG-MN-4.01-1008, as stated in the original question justification.

Per CNG-MN-4.01-1008, preconditioning is defined as "The unacceptable alteration, variation, manipulation or adjustment of the physical condition of a System, Structure, or Component (SSC) before Technical Specification surveillance... Such testing can alter one or more operational parameters which results in acceptable test results that can mask the actual as-found condition of the SSC..."

Choice D in the question states, "Starting and running Core Spray pump 112 to bring it up to operating temperature." In certain surveillances, this would be considered preconditioning. For example, N1-ST-M1A, Liquid Poison Pump 11 Operability Test, requires data to be taken immediately upon pump start. However, N1-ST-Q1C section 2.2.16 states the following: "Core Spray and Core Spray Topping Pump testing data is taken after the pumps have run for a minimum of two minutes under stabilized conditions."

Conclusions:

The specific surveillance listed in the question stem directs starting the pump, making system manipulations, and allowing conditions to be stable for a minimum of 2 minutes prior to taking the required readings for the surveillance. Since allowing conditions to stabilize is directed by this surveillance, it cannot be considered pre-conditioning. This is equivalent to the wording of Choice D. Therefore, both B and D are correct answers.

Recommendation:

Recommend accepting both choices B and D as correct.

NRC Response

Comment accepted. Typically, surveillances require test data to be taken immediately upon pump start. However, the Core Spray Pump and Valve Operability Test procedure does not contain this direction. In this test, the pump is started and run until conditions stabilize: Per Discussion Step 2.2.16, "... pump testing data is taken after the pumps have been run for a minimum of two minutes under stabilized conditions." This procedure direction matches the action described in choice D. Therefore, in addition to choice B, choice D is also correct.

Question 84
Part 1 - Original Question

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295032 2.4.18
	Importance Rating	4.0

High Secondary Containment Area Temperature

Knowledge of the specific bases for EOPs.

Proposed Question: #84

The plant has scrammed from 100% power with the following:

- A steam leak has developed from the Reactor Water Cleanup (RWCU) system.
- Attempts to isolate RWCU have been unsuccessful.
- N1-EOP-2, RPV Control, and N1-EOP-5, Secondary Containment Control, have been entered.
- An Operator in the field reports:
  - Reactor Building 261' West temperature is 136°F and rising slowly.
  - Reactor Building 261' East temperature is 145°F and rising slowly.
  - Reactor Building 281' West temperature is 110°F and rising slowly.
  - Reactor Building 281' East temperature is 119°F and rising slowly.

Which one of the following describes the required action and the basis for the associated temperature limit, in accordance with N1-EOP-2 and/or N1-EOP-5?

	Required Action	Basis for Temperature Limit
A.	Perform an RPV Blowdown	Personnel access
B.	Perform an RPV Blowdown	Reactor Building design limit
C.	Rapidly depressurize using Turbine Bypass Valves	Personnel access
D.	Rapidly depressurize using Turbine Bypass Valves	Reactor Building design limit

Proposed Answer: A

Explanation: The given conditions indicate that a primary system is discharging into the Secondary Containment, the system cannot be isolated, and two General Areas (RB 261' West and East) have exceeded the Maximum Safe Temperature of 135°F. With two General Areas above the Maximum Safe Temperature, an RPV Blowdown is required. The basis for the 135°F Maximum Safe Temperature limit is to allow personnel access into the Secondary Containment to perform safe shutdown actions.

B. Plausible – These temperatures are well above normal Reactor Building temperatures, however the basis for the EOP Maximum Safe Temperature is either personnel access (limiting at Nine Mile Point Unit 1) or equipment operability, not Reactor Building design temperature.
C. Plausible – If only one General Area temperature was above Maximum Safe with a second General Area trending towards the limit, then RPV Blowdown would not yet be required and rapid depressurization with Turbine Bypass Valves would be correct.
D. Plausible – If only one General Area temperature was above Maximum Safe with a second General Area trending towards the limit, then RPV Blowdown would not yet be required and rapid depressurization with Turbine Bypass Valves would be correct. These temperatures are well above normal Reactor Building temperatures, however the basis for the EOP Maximum Safe Temperature is either personnel access (limiting at Nine Mile Point Unit 1) or equipment operability, not Reactor Building design temperature.

Technical Reference(s): N1-EOP-5, N1-EOP-2, NER-1M-095, GAI-OPS-20

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Question 84
Part 2 - Justification

Analysis:

The question presents a situation where plant parameters are exceeding the Maximum Safe Value for Reactor Building area temperature. The first half of the question requires selecting the appropriate action for the given conditions. This portion of the question is correct as originally keyed. The second half of the question requires selecting the basis for the Maximum Safe temperature.

The EOP bases state, "The 'Maximum Safe Value' of a parameter 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."

The bases go on to state, "A parameter above the Maximum Safe Value in two separate areas is indicative of a widespread problem posing a direct and immediate threat to secondary containment, equipment in the secondary containment and safe operation of the plant."

The second half of the question provides the following choices:

- Reactor Building design limit (originally keyed as incorrect)
- Personnel access (originally keyed as correct)

"Personnel access" is still a correct answer for the question. "Reactor Building design limit" is also a correct answer. The wording of "Reactor Building design limit" is sufficiently open to interpretation to fit within the EOP bases for Maximum Safe temperature. Reactor Building safety analyses provide an assumed limit on the highest temperature expected to persist in generalized areas of the Reactor Building (150°F). This provides an upper limit on the operating conditions for equipment in the Reactor Building that is needed for safe shutdown.

SDBD-601 provides the following:

2.7.2.1 Requirement - As a minimum, the Reactor Building HVAC System shall be designed for the following pressures and temperatures.

Subsystem	Design Pressure	Design Temperature
202	(OI-S-601-018)	(OI-S-601-018)
202.1	50 psig	200°F
201.2	35 psig	150°F

Basis for the Requirement - The design basis for the design temperatures and pressures of Subsystems 202, 202.1 and 201.2 were established by NMPC design personnel based on expected maximum operating parameters. Specifically, for 201.2 piping, the Emergency Ventilation System was expected to be used for nitrogen purge operation if the primary containment environment has high levels of contaminants. It may also be used as a Post-LOCA vent path, via the Post-LOCA vent lines. For these reasons, the Emergency Ventilation System piping from the Normal Ventilation exhaust piping to the Duct work downstream of the Emergency Ventilation Fans was designed to 201.2 requirements. Design conditions for this

PSRS were established as 35 psig and 150°F to envelope potential containment environment parameters.

The same 150°F temperature limit is given in section 2.7.3.1 for Emergency Ventilation and Normal ventilation ducting. The temperature is based on the worst case accident for the secondary containment, a break in the Emergency Cooling System.

Taking action using a Maximum Safe value of 135°F limits damage to equipment inside the Reactor Building that is required for safe shutdown of the plant.

<u>Conclusions:</u>

Both choices in the second half of the question are correct per EOP bases.

<u>Recommendation:</u>

Accept both choices A and B as correct.

<u>NRC Response</u>

Recommendation accepted. The basis for the Maximum Safe Value of a parameter (here, 135F reactor building temperature), is the highest value at which, among other considerations, personnel can perform any actions necessary for the safe shutdown of the plant. "Personnel access" is therefore a correct choice for Basis for Temperature Limit. However, the bases also state a parameter above the MSV ... is indicative of a widespread problem posing a direct and immediate threat to secondary containment, i.e., the reactor building. (At NMP1, secondary containment is synonymous with reactor building.) Stem conditions indicate temperatures are above the MSV, so there is a direct and immediate threat to the reactor building. The threat could logically be to the reactor building design limit. Thus, Reactor Building design limit is also a correct choice for Basis for Temperature Limit. In conclusion, A and B are both correct.

Question 95
Part 1 - Original Question

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.1
	Importance Rating	4.4

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: #95

Preparations for a plant startup are in progress with the following:

- Based on plant conditions, one step in N1-OP-43A, Plant Startup, Section E.1.0, Plant Startup Preparations and Prerequisites, CANNOT be performed and is NOT required to be performed.
- All other steps in N1-OP-43A Section E.1.0 are complete and signed off.
- It is desired to continue on to N1-OP-43A Section E.2.0, Approach to Criticality and Vessel Heatup.

Which one of the following describes the required action to proceed with the startup in accordance with N1-OP-43A?

A. Record the step on a Variance Form. Only one SRO is required to review and approve the variance.

B. Record the step on a Variance Form. Two SROs are required to review and approve the variance.

C. Process a Technical Procedure Step Deletion Screening Form. Only one SRO is required to review and approve the step deletion.

D. Process a Technical Procedure Step Deletion Screening Form. Two SROs are required to review and approve the step deletion.

Proposed Answer: B

Explanation: N1-OP-43A contains specific guidance in section C.1.0 regarding processing of a step that cannot be performed and is not required to be performed. The procedure requires recording the step on a Variance Form located as an attachment in N1-OP-43A. Both the Control Room Supervisor and Shift Manager (2 SROs) must sign the variance form entry.

A. Plausible – Both the Control Room Supervisor and Shift Manager must sign the variance form entry, therefore it is two SROs, not just one, that must review and approve the variance.
C. Plausible – CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form, is the normal method for such a situation with a technical procedure. However, a plant startup is a unique situation where the technical procedure contains its own method for tracking such steps. Both the Control Room Supervisor and Shift Manager must sign the variance form entry, therefore it is two SROs, not just one, that must review and approve the variance.
D. Plausible – CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form, is the normal method for such a situation with a technical procedure. However, a plant startup is a unique situation where the technical procedure contains its own method for tracking such steps.

Technical Reference(s): N1-OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-OP-43A-CT-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Question 95
Part 2 - Justification

Analysis:

The stem conditions present a reactor startup with a prerequisite step that cannot be performed and asks the process for that situation.

N1-OP-43A, Plant Startup, is unique in that it contains internal guidance for how to proceed when a step cannot be performed. The keyed correct answer, B, is correct in accordance with N1-OP-43A section C.1.0.

However, the normal method for such a situation with a technical procedure is to use CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form. Nothing in the stem or answer choices would prohibit the use of the normal step deletion process, which makes choice D also correct.

Conclusions:

If an operator used CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form, in this situation, they would not be wrong. Either process would be acceptable for deleting the step.

Recommendation:

Recommend accepting both choices B and D as correct.

NRC Response

Comment accepted. The Variance Form contained in N1-OP-43A documents situations that prevent a particular step from being signed off. However, the normal method to address a procedure step that cannot be performed is found in CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form. This procedure accomplishes the same goal as OP-43A, in fact with more rigor than required by OP-43A. Furthermore, the critical safety review required in both procedures - two SROs must review and approve the variance - is consistent. Either process is acceptable for deleting a step that cannot be performed. Therefore, both B and D choices are correct.

Question 99
Part 1 - Original Question

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.38
	Importance Rating	4.4

Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Proposed Question: #99

An emergency is in progress and you are the Shift Manager / Emergency Director (SM/ED).

Given the following responsibilities:

(1) Determining the necessity for a local area/building evacuation
(2) Authorizing emergency workers to exceed normal radiation exposure limits
(3) Making the decision to notify off-site emergency management agencies
(4) Making Protective Action Recommendations (PARs) to off-site emergency management agencies

Which one of the following identifies which of these responsibilities shall NOT be delegated, in accordance with the Site Emergency Plan?

A. (4) only

B. (2) and (3) only

C. (2), (3), and (4) only

D. (1), (2), (3), and (4)

Proposed Answer: C

Explanation: The Site Emergency Plan lists the following responsibilities of the SM/ED that shall NOT be delegated:

- Classification and declaration of the emergency event as an Unusual Event, Alert, Site Area Emergency or General Emergency.
- Determining the necessity for an **exclusion** area evacuation.
- Authorizing emergency workers to exceed normal radiation exposure limits.
- Making the decision to notify off-site emergency management agencies.
- Making Protective Action Recommendations (PARs) as necessary to offsite emergency management agencies.

Items (2), (3), and (4) correspond to three of these bullets. Item (1) is similar to the second bullet, but not the same. The need for local area/building evacuations may be determined by other members of the operating crew, such as the CRS or an RO carrying out an SOP with evacuation requirements.

A. Plausible – Item (4) shall NOT be delegated, and is one of the more severe items given. However, items (2) and (3) also shall NOT be delegated.
B. Plausible – Items (2) and (3) shall NOT be delegated, and are two of the more severe items given. However, item (4) also shall NOT be delegated.
D. Plausible – Items (2), (3), and (4) shall NOT be delegated, however item (1) may be performed by other members of the operating crew.

Technical Reference(s): Site Emergency Plan section 5.2

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPL000-TO

Question Source: Bank – NMP2 2014 Audit #100

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Question 99
Part 2 - Justification

Analysis:

The stem conditions ask the applicant to determine what actions the SM/ED cannot delegate.

Based on a revision to the Site Emergency plan, an error exists in the original justification when it states "Determining the necessity for an **exclusion** area evacuation" in non-delegable. This is the wording from a previous revision of the Site Emergency Plan and led to keying choice C as correct.

The current Site Emergency Plan, which the candidates were trained on, states that "ensuring appropriate evacuation actions for plant personnel" is a non-delegable action. This statement encompasses (1) Determining the necessity for a local area/building evacuation. Therefore, all listed responsibilities are non-delegable in accordance with the Site Emergency Plan.

Conclusions:

The originally accepted answer, C, is not technically valid.

Recommendation:

Recommend re-keying correct answer to choice D.

NRC Response

Comment accepted. The applicants were trained on a version of the Site Emergency Plan that lists actions the SM/ED cannot delegate. An action that shall not be delegated is to ensure appropriate evacuation actions for plant personnel, per the Shift Manager Checklist. That statement encompasses the first responsibility listed in the question conditions. Therefore, the correct answer is changed to D.