Question #4 - Justification for Accepting Two Correct Answers

Choice 'A' is correct because it describes the basis for the 60 sec generator trip time delay explained in the second bullet of P-12, Electrical *Systems Precautions, Limitations, and Setpoints*, step 4.1.5. The second bullet of P-12 step 4.1.5 states the following:

"On a major loss of coolant accident (double-ended shear of the reactor coolant cold leg), as the coolant rushes out of the break, the RCP impeller, shaft, flywheel, etc., can be oversped. RCP overspeed could cause catastrophic failure of the flywheel resulting in missiles which could damage the containment liner or ECCS components within the containment. The RCP overspeed concern is minimized by locking the RCPs at -60 Hz for the duration (60 seconds) of the generator trip time delay."

Choice 'C' is also correct because it describes the basis for the 60 sec generator trip time delay explained in the first bullet of P-12 step 4.1.5. The first bullet of P-12 step 4.1.5 states the following:

"An immediate turbine trip generator trip coincident with a failure of automatic Bus transfer or electrical Buses failure could result in a loss of forced reactor coolant flow. If the reactor trips due to overpower, over-temperature, or low pressure conditions, the loss of flow could make the consequences of the accident more severe than reported in the UFSAR. However, if pumping power is lost with a time delayed generator trip, loss of flow is not considered serious because the reactor has been shut down for a period of time."

UFSAR 7.2.2.2.13 states the following:

"Turbine trip causing a reactor trip is provided to anticipate probable plant transients and to avoid the resulting thermal transients. If the reactor were not tripped by the turbine trip, the overtemperature delta T or high pressure trip would prevent reactor safety limits from being exceeded."

Additionally, Technical Specification basis for Reactor Trip System Instrumentation B.3.3.1 states the following relative to the Overtemperature Delta T reactor trip:

"The Overtemperature ΔT trip Function is provided to ensure that the design limit departure from nucleate boiling ratio (DNBR) is met ... The Overtemperature ΔT trip Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase."

As stated in the explanation above, if an immediate loss of flow occurs when the reactor trips on over-temperature the consequences of the accident can be more severe than reported in the UFSAR. The consequences of the accident are more severe because of power-to-flow concerns, i.e. power is higher with no forced RCS flow sooner than with the 60 second generator trip time delay.

Therefore, choice 'C' adequately describes the basis for the 60 sec generator trip time delay explained in the first bullet of P-12 step 4.1.5.

Examination Outline Cross-reference:

Proposed Re-write

Level	RO		SRO
Tier #	3		
Group #	1		
K/A #	G1	2.1.32	
Importance Rating	3.8		

Conduct of Operations - Ability to explain and apply system limits and precautions.

RO Question # 4 Rev 1

Which one of the following statements describes a basis, as explained in P-12, ELECTRICAL SYSTEMS PRECAUTIONS, LIMITATIONS, AND SETPOINTS, for why the generator trip circuit is designed to be time-delayed, such that the generator trip occurs later than the turbine trip on most turbine trips?

A. On a Large Break LOCA the RCP can overspeed causing the motor flywheel to become a missile hazard which could damage the containment liner or ECCS components in containment.

B. On a Large Break LOCA the RCP can overspeed causing the RCP impeller to become a missile hazard which could damage the containment liner or ECCS components in containment.

C. On a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds to prevent formation of excessive voids in reactor head upon the reactor trip.

D. On a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds to prevent an RCS pressure transient upon reactor trip.

Proposed Answer: A

Explanation (Optional):

A. Correct. Per P-12, on a major loss of coolant accident, the RCP impeller, shaft, flywheel, etc., can overspeed. RCP overspeed could cause catastrophic failure of the flywheel resulting in missiles which could damage the containment liner or ECCS components within containment.

B. Incorrect. Plausible because it is very similar to the correct answer. Incorrect because it identifies the RCP impeller as the missile hazard.

C. Incorrect. Plausible because on natural circulation cooldown and depressurization, potential for void formation may occur. Incorrect because the RCP's remain running after sixty seconds since they transfer to off-site power.

D. Incorrect. Plausible because it is very similar to the other basis for this time delay. Incorrect because the reactor trip involved has to be a reactor trip that provides DNB protection. The reactor trip from turbine trip does not provide DNB protection.

Technical Reference(s): P-12
P-1

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0501C, 1.13

Question Source: Bank # C062.0053

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
KA Administered	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.32	
	Importance Rating	3.8		

Conduct of Operations - Ability to explain and apply system limits and precautions.

RO Question # 4

Which one of the following statements describes a basis, as explained in P-12, ELECTRICAL SYSTEMS PRECAUTIONS, LIMITATIONS, AND SETPOINTS, for why the generator trip circuit is designed to be time-delayed, such that the generator trip occurs later than the turbine trip on most turbine trips?

A. On a Large Break LOCA the RCP can overspeed causing the motor flywheel to become a missile hazard which could damage the containment liner or ECCS components in containment.

B. On a Large Break LOCA the RCP can overspeed causing the RCP impeller to become a missile hazard which could damage the containment liner or ECCS components in containment.

C. On a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds to prevent a power-to-flow concern upon reactor trip.

D. On a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds to prevent an RCS pressure transient upon reactor trip.

Proposed Answer: A

Explanation (Optional):

A. Correct. Per P-12, on a major loss of coolant accident, the RCP impeller, shaft, flywheel, etc., can overspeed. RCP overspeed could cause catastrophic failure of the flywheel resulting in missiles which could damage the containment liner or ECCS components within containment.

B. Incorrect. Plausible because it is very similar to the correct answer. Incorrect because it identifies the RCP impeller as the missile hazard.

C. Incorrect. Plausible because it is very similar to the other basis for this time delay. Incorrect because the reactor trip involved has to be a reactor trip that provides DNB protection. The reactor trip from turbine trip does not provide DNB protection.

D. Incorrect. Plausible because it is very similar to the other basis for this time delay. Incorrect because the reactor trip involved has to be a reactor trip that provides DNB protection. The reactor trip from turbine trip does not provide DNB protection.

Technical Reference(s): P-12 P-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R0501C, 1.13

Question Source: Bank # C062.0053

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

4.0 PRECAUTIONS AND LIMITATIONS

4.1. Main Generation

4.1.1. The Main Generation System is designed to produce electrical power at 19.5kV and transmit this power off-site to the transmission system (normally 34.5kV or 115kV). The main generator is rated for 613.5 MWe (gross) output at a voltage of 19.5kV. The main transformer steps this voltage up to 115kV for distribution through the generating system.

4.1.2. The normal source of auxiliary power during plant operation is the main generator output via the Unit Auxiliary Transformer 11. The Station Auxiliary Transformers 12A and 12B step down 34.5kV from lines entering the station to 4160V for use in the station 4160V and 480V Electrical Systems.

4.1.3. Standby power required during plant startup, shutdown, and after a reactor trip is supplied from the Station Auxiliary Transformers 12A and 12B.

- Transformer 12A is supplied via circuit 7T which originates at Substation 13A. Transformer 7 in substation 13A steps down voltage from 115kV to 34.5kV to supply circuit 7T.

- Transformer 12B is supplied via circuit 767 which originates in substation 13A. Transformer 6 in substation 13A steps down voltage from 115kV to 34.5kV to supply circuit 767.

4.1.4. Any type of fault condition that can occur within the 19.5kV system will cause a generator trip. This will result in the de-energization of the Unit Auxiliary Transformer 11 and the tripping of the generator output Breakers 1G13A72 and 9X13A72. Bus tie Breakers from 4160 volt Bus 12B to 11B and from Bus 12A to 11A will automatically close to re-energize these Buses with power. Attachment 3, Generator Trips, lists trips.

4.1.5. On most turbine trips, generator trip circuit is delayed approximately 60 seconds. The following are two safety-related bases for generator trip time delay:

- An immediate turbine trip generator trip coincident with a failure of automatic Bus transfer or electrical Buses failure could result in a loss of forced reactor coolant flow. If the reactor trips due to overpower, over-temperature, or low pressure conditions, the loss of flow could make the consequences of the accident more severe than reported in the UFSAR. However, if pumping power is lost with a time delayed generator trip, loss of flow is not considered serious because the reactor has been shut down for a period of time.

- On a major loss of coolant accident (double-ended shear of the reactor coolant cold leg), as the coolant rushes out of the break, the RCP impeller, shaft, flywheel, etc., can be oversped. RCP overspeed could cause catastrophic failure of the flywheel resulting in missiles which could damage the containment liner or ECCS components within the containment. The RCP overspeed concern is minimized by locking the RCPs at -60 Hz for the duration (60 seconds) of the generator trip time delay.

will directly trip the reactor to prevent departure from nucleate boiling. This trip is bypassed below 8% power by permissive P-7.

The underfrequency on the pump power supply trip provides reactor protection following a major grid frequency disturbance. If an underfrequency condition below 57.7 Hz (one-out-of-two logic) exists on both reactor coolant pump buses, all reactor coolant pump breakers and the reactor are tripped. This is done because an underfrequency condition will slow down the pumps thereby reducing their coastdown time following a pump trip.

The undervoltage and underfrequency trip logic is shown in Drawing 33013-1353, Sheet 4.

7.2.2.2.12 Safety Injection System Actuation Trip

A reactor trip occurs on the actuation of the safety injection system. The means of actuating the safety injection system trips are described in Section 7.3.2.

7.2.2.2.13 Turbine Trip/Reactor Trip

Turbine trip causing a reactor trip is provided to anticipate probable plant transients and to avoid the resulting thermal transients. If the reactor were not tripped by the turbine trip, the overtemperature delta T or high pressure trip would prevent reactor safety limits from being exceeded. By utilizing this trip, undesirable excursions are prevented rather than terminated.

The trip is sensed by a decrease in emergency trip system oil pressure or all stop valves shut. Three switches are mounted on the emergency trip oil header and their outputs are tied together in a two-out-of-three logic. This logic will initiate a reactor trip (auto-stop oil pressure less than 45 psig) provided the reactor is operating above 50% power as sensed by permissive P-9. It is not necessary to trip the reactor if it is operating below 50% power since rod control in conjunction with steam dump can accomodate a 50% load rejection without a reactor trip (Section 10.7.1). Turbine trip leading to reactor trip logic is shown in Drawing 33013-1353, Sheet 3.

7.2.2.2.14 Low-Low Steam-Generator Water Level Trip

The purpose of this trip is to protect the steam generators for the case of a sustained steam/ feedwater flow mismatch. The trip is actuated on two-out-of-three low-low water level signals in either steam generator. The trip logic is shown in Drawing 33013-1353, Sheet 13.

7.2.2.3 Interlocks

A number of reactor trips applicable to power range operation are automatically bypassed to permit reactor startup and low power operation. The following trip functions are blocked by a coincidence of three-out-of-four power range nuclear flux channels reading less than 8% power and one-out-of-two low turbine load (turbine impulse chamber pressure) signals:

A. Low reactor coolant flow (both loops).

B. Reactor coolant pump breaker trip (both loops).

C. Turbine trip with P-9 permissive present.

D. Undervoltage.

In MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods are not fully inserted, the Source Range Neutron Flux trip Function must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal and all rods are fully inserted, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit departure from nucleate boiling ratio (DNBR) is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, T_{avg}, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses the ΔT of each loop as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution $f(\Delta I)$ - the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

Question #26 - Justification for Accepting Two Correct Answers Rev. 1

Both choice 'B' and choice 'D' are correct because the statement in the second part of question #26 does not clearly ask which design basis accident results in the highest peak containment pressure.

The second part of Question #26 states the following:

"The design basis accident for the peak pressure limit in Containment is ___________."

Technical Specification Basis B 3.6.4 for Containment Pressure states the following:

"Containment internal pressure is an initial condition used in the DBA analyses performed to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The worst case SLB generates larger mass and energy releases than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1)."

Both choice 'B' and choice 'D' are correct because as stated above both the LOCA and SLB are limiting DBAs considered relative to containment pressure. It is true that the SLB accident produces the highest containment pressure, but this is not clearly asked for in part #2 of the question. For SLB to be the only correct answer then part #2 should have simply asked "which accident produces the highest pressure in containment?" The original wording is confusing thus resulting in candidates selecting LOCA vice SLB because both are considered, and LOCA is the overall limiting design basis accident for containment based on offsite dose. Therefore, both choices 'B' and 'D' should be accepted as correct.

Examination Outline Cross-reference:	Level	RO		SRO
Proposed Re-write	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.42	
	Importance Rating	3.9		

Ability to recognize system parameters that are entry-level conditions for Technical Specifications

RO Question # 26 Rev 1

The crew has placed the Containment Mini-Purge system in service and notes that Containment Pressure is 0.4 psig and rising slowly.

If pressure continues to rise, the crew will be required to enter a Tech Spec Action statement at __(1)__ psig, which is the initial pressure used in the analysis for determining the peak pressure limit. The design basis accident resulting in the highest peak pressure in Containment is a __(2)__ .

A. (1) 0.5 psig; (2) Steamline break inside CNMT

B. (1) 1.0 psig; (2) Steamline break inside CNMT

C. (1) 0.5 psig; (2) LOCA

D. (1) 1.0 psig; (2) LOCA

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the value in (1) is the MCB alarm setpoint which would require CNMT depressurization while (2) is the correct accident.

B. Correct. Per ITS 3.6.4 basis, the initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case steamline break, 59.6 psig, does not exceed the containment design pressure of 60 psig.

C. Incorrect. Plausible because the value in (1) is the MCB alarm setpoint which would require CNMT depressurization, while (2) is plausible because peak CNMT pressure following DBA LOCA is a valid concern (but not after EPU).

D. Incorrect. Plausible because (1) is the correct setpoint and (2) is plausible because peak CNMT pressure following DBA LOCA is a valid concern (but not after EPU).

Technical Reference(s):	ITS Basis B3.6.4	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2101C, 1.12 and 1.13

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
As Administered	Tier #	3	
	Group #	2	
	K/A #	G2	2.2.42
	Importance Rating	3.9	

Ability to recognize system parameters that are entry-level conditions for Technical Specifications

RO Question # 26

The crew has placed the Containment Mini-Purge system in service and notes that Containment Pressure is 0.4 psig and rising slowly.

If pressure continues to rise, the crew will be required to enter a Tech Spec Action statement at __(1)__ psig, which is the initial pressure used in the analysis for determining the peak pressure limit. The design basis accident for the peak pressure limit in Containment is __(2)__ .

A. (1) 0.5 psig; (2) Steamline break inside CNMT

B. (1) 1.0 psig; (2) Steamline break inside CNMT

C. (1) 0.5 psig; (2) LOCA

D. (1) 1.0 psig; (2) LOCA

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the value in (1) is the MCB alarm setpoint which would require CNMT depressurization while (2) is the correct accident.

B. Correct. Per ITS 3.6.4 basis, the initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case steamline break, 59.6 psig, does not exceed the containment design pressure of 60 psig.

C. Incorrect. Plausible because the value in (1) is the MCB alarm setpoint which would require CNMT depressurization, while (2) is plausible because peak CNMT pressure following DBA LOCA is a valid concern (but not after EPU).

D. Incorrect. Plausible because (1) is the correct setpoint and (2) is plausible because peak CNMT pressure following DBA LOCA is a valid concern (but not after EPU).

Technical Reference(s):	ITS Basis B3.6.4	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: R2101C, 1.12 and 1.13

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Comments:

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, post accident containment pressures could exceed calculated values. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment pressure outside the limits of the LCO violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses performed to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The worst case SLB generates larger mass and energy releases than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case SLB, 59.6 psig, does not exceed the containment design pressure, 60 psig.

The containment was also designed for an external pressure load equivalent to -2.5 psig. However, internal pressure is limited to -2.0 psig based on concerns related to providing continued cooling for the reactor coolant pump motors inside containment.

Question #31 - Justification for Accepting Two Correct Answers Rev.2

Question #31 states the following:

"Given the following:

- The plant is at full power.
- Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW, is lit.
- One SW pump is running.

Per the alarm response, Annunciator C-10 alarms when Service Water flow from any CNMT Recirc Fan is less than ___(1)___ gpm and either CNMT Recirc Fan(s) service water outlet (FCV-4561/FCV-4562) is full open; and, with only a single service water pump operating, **refer to** ___(2)___."

Only choices 'C' and 'D' are plausible because Alarm Response Procedure AR-C-10 states the low flow setpoint is 1050 gpm.

The second part of question #31 requires the candidate to determine which Service Water AP should be referenced based on the conditions stated in the stem of the question. Annunciator C-10 is listed as a possible symptom in both AP-SW.1 and AP-SW.2 and is therefore a possible indication of either loss of SW pumps or a SW leak or both. Alarm Response Procedure AR-C-10 has steps to refer to AP-SW.2 if the alarm is due to loss of SW pumps and to refer to AP-SW.1 if a SW leak is indicated.

As written, the stem of the question does not contain sufficient information without making assumptions to allow a candidate to determine positively whether alarm is due to "loss of SW pumps" or if a "SW leak is indicated." Simply stating that one SW pump is running doesn't necessarily imply that alarm is due to loss of SW pumps or that a SW leak is not indicated.

Additionally, the question is asking which SW AP to "refer to". Refer to simply denotes a procedure which may provide necessary or useful information. With only the conditions stated in the stem of the question, it would not be unreasonable to expect an operator with a healthy questioning attitude to reference both APs to determine the optimal recovery actions.

In summary, either choice 'C' or choice 'D' should be considered correct since AP-SW.1 and AP-SW.2 are both referenced in AR-C-10. Without being able to definitely determine the cause of the alarm, it would be expected that both APs should be referenced.

Examination Outline Cross-reference:

Proposed Re-write

Level	RO		SRO
Tier #	2		
Group #	1		
K/A #	022	2.4.31	
Importance Rating	4.2		

Knowledge of annunciator alarms, indications, or response procedures. (Regarding Containment Cooling)

RO Question # 31 Rev 1

Given the following:

- The plant is at full power.
- Annunciator J-9, SAFEGUARD BREAKER TRIP, is lit
- Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW, is lit.
- One of the two running SW pump trips.

Per the alarm response, annunciator C-10 alarms when Service Water flow from any CNMT Recirc Fan is less than __(1)__ gpm and either CNMT Recirc Fan(s) service water outlet (FCV-4561/FCV-4562) is full open; and, with only a single service water pump operating, refer to __(2)__ .

	(1)	(2)
A.	1100	AP-SW.1, Service Water Leak
B.	1100	AP-SW.2, Loss of Service Water
C.	1050	AP-SW.1, Service Water Leak
D.	1050	AP-SW.2, Loss of Service Water

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the examinee can easily confuse alarm C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM with the setpoint of alarm K-21, SFP LOW FLOW , which is 1100 gpm. Part 2 is plausible because license class students are always challenged to differentiate entry to AP-SW.1 versus AP-SW.2. Additionally, both AP-SW.1 and AP-SW.2 verify at least one SW pump running in each loop. Incorrect because C-10 alarms when flow is < 1050 gpm, and the appropriate procedure for a single pump running is AP-SW.2.

B. Incorrect. Plausible because the examinee can easily confuse alarm C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM with the

setpoint of alarm K-21, SFP LOW FLOW , which is 1100 gpm, and the second part is correct. Incorrect because C-10 alarms when flow is < 1050 gpm.

C. Incorrect. Plausible because the first part is correct, and license class students are always challenged to differentiate entry to AP-SW.1 versus AP-SW.2. Both AP-SW.1 and AP-SW.2 are referred to in the required actions section. Additionally, both AP-SW.1 and AP-SW.2 verify at least one SW pump running in each loop. Incorrect because the appropriate procedure for a single pump running is AP-SW.2.

D. Correct. Per the Alarm Response, the alarm setpoint is < 1050 gpm, and the correct procedure is AP-SW.2.

Technical Reference(s): AR-C-10

Proposed References to be provided to applicants during examination: None

Learning Objective: R5101C 1.04

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

Comments: Administratively, the plant cannot operate at full power with only a single service water pump running. The question states: "if only a single service water pump is operating". This infers that one or more service water pumps must have tripped. There is no information suggesting that a service water leak exists. With the lack of specifics, the examinee cannot assume that a leak exists. Therefore, the appropriate procedure must be AP-SW.2. The examinee must use system knowledge to determine what would cause the alarm, and recognize the purpose of the AP-SW procedures to select the appropriate procedure. Just recognizing the purpose makes this an RO question rather than an SRO only question.

Examination Outline Cross-reference:	Level	RO	SRO
As Administered	Tier #	2	
	Group #	1	
	K/A #	022	2.4.31
	Importance Rating	4.2	

Knowledge of annunciator alarms, indications, or response procedures. (Regarding Containment Cooling)

RO Question # 31

Given the following:

- The plant is at full power.
- Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW, is lit.
- One SW pump is running.

Per the alarm response, annunciator C-10 alarms when Service Water flow from any CNMT Recirc Fan is less than __(1)__ gpm and either CNMT Recirc Fan(s) service water outlet (FCV-4561/FCV-4562) is full open; and, with only a single service water pump operating, refer to __(2)__ .

	(1)	(2)
A.	1100	AP-SW.1, Service Water Leak
B.	1100	AP-SW.2, Loss of Service Water
C.	1050	AP-SW.1, Service Water Leak
D.	1050	AP-SW.2, Loss of Service Water

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because the examinee can easily confuse alarm C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM with the setpoint of alarm K-21, SFP LOW FLOW , which is 1100 gpm. Part 2 is plausible because license class students are always challenged to differentiate entry to AP-SW.1 versus AP-SW.2. Additionally, both AP-SW.1 and AP-SW.2 verify at least one SW pump running in each loop. Incorrect because C-10 alarms when flow is < 1050 gpm, and the appropriate procedure for a single pump running is AP-SW.2.

B. Incorrect. Plausible because the examinee can easily confuse alarm C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM with the setpoint of alarm K-21, SFP LOW FLOW , which is 1100 gpm, and the second part is

correct. Incorrect because C-10 alarms when flow is < 1050 gpm.

C. Incorrect. Plausible because the first part is correct, and license class students are always challenged to differentiate entry to AP-SW.1 versus AP-SW.2. Both AP-SW.1 and AP-SW.2 are referred to in the required actions section. Additionally, both AP-SW.1 and AP-SW.2 verify at least one SW pump running in each loop. Incorrect because the appropriate procedure for a single pump running is AP-SW.2.

D. Correct. Per the Alarm Response, the alarm setpoint is < 1050 gpm, and the correct procedure is AP-SW.2.

Technical Reference(s): AR-C-10

Proposed References to be provided to applicants during examination: None

Learning Objective: R5101C 1.04

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

Comments: Administratively, the plant cannot operate at full power with only a single service water pump running. The question states: "if only a single service water pump is operating". This infers that one or more service water pumps must have tripped. There is no information suggesting that a service water leak exists. With the lack of specifics, the examinee cannot assume that a leak exists. Therefore, the appropriate procedure must be AP-SW.2. The examinee must use system knowledge to determine what would cause the alarm, and recognize the purpose of the AP-SW procedures to select the appropriate procedure. Just recognizing the purpose makes this an RO question rather than an SRO only question.

SS. **Perform**

1. To carry through to completion
   - Example - Perform the following.

TT. **Place**

1. To move a control to a specific position.
   - Example - Place in auto.

UU. **Raise**

1. To increase in value or amount.
   - Example - Raise charging flow to restore PRZR level

VV. **Record**

1. To document in writing
   - Example - Record RCS pressure.

WW. **Refer to**

1. To utilize a procedure, attachment, or document to address concerns or conditions
   - Example - Refer to AP-IA.1 Loss of Instrument Air

XX. **Reset**

1. To restore to a previous or initial state. Generally directs placement of a component or control to a pre-trip or ready/standby condition.
   - Example - Reset SI.

YY. **Restore**

1. To place in an original condition.
   - Example - Restore power to any train of AC emergency busses.

ZZ. **Return to**

1. To transition to a previous step within the same procedure, or to a previous procedure
   - Example - Return to step 2.
   - Return to procedure and step in effect.

AAA. **Ruptured**

1. Condition in which a steam generator has primary to secondary leakage in excess of charging pump capacity such that SI is (or was) required to maintain RCS inventory.

BBB. **Sample**

1. To take a representative portion for the purpose of examination
   - Example - Sample steam generator blowdown for activity.

2. The word OR is used between alternative conditions. Use of the word OR implies the inclusive sense. This application may also use the term AND/OR. The exclusive sense of the word OR is denoted by using the terminology; either A OR B, but not both.

3. When two or more actions or criteria are separated by an OR condition, only one action needs to be successfully taken or one criteria successfully met to allow progression to the next step or sub step.

4. Action steps contingent upon certain conditions or combinations of conditions, begin with the logic terms IF or WHEN followed by a description of the condition(s), a comma, the logic term THEN and the action to be taken. IF is used for unexpected or possible conditions, WHEN is used for expected or probable conditions, and THEN is only used in conditional statements.

M. **Use of Reference Terms**

1. The words “go to” followed by only a step number directs transition to a subsequent step within the procedure being used.

2. The words “return to” followed by only a step number directs transition to a previous step within the procedure being used.

3. The words “go to” followed by a procedure designator and title and a step number, direct a transition to the specified EOP/AP. If no step number is specified, then transition is to the beginning of a specified procedure.

   - EXAMPLE: Go to ES-0.1, Reactor Trip Response, Step 20.

4. The words “refer to” followed by a procedure designator and title, are used to denote a procedure which may provide necessary or useful information during the execution of an EOP/AP. In general, those procedures referenced cover low probability occurrences, or plant evaluations with their own procedures whose inclusion in the EOP/AP would cause excessive complication of and reduced effectiveness of the EOP/AP. Referenced procedures should be performed in parallel with the primary procedure.

   - Example: Refer to ER-AFW.1, Alternate Water Supply to AFW Pumps.

5. Procedures entered for supplemental guidance or from CSFST direction may contain a “return to” statement.

6. A procedure with multiple entry conditions may state: “RETURN TO PROCEDURE AND STEP IN EFFECT”, which denotes a return to the last previous EOP and step in use.

7. If awaiting a condition to be satisfied before performing the actions in a step/substep, then the RNO may direct the operator to continue with subsequent steps with the stipulation that when the desired condition is satisfied, the bypassed actions should be performed. The word “DO” followed by the appropriate step/sub step numbers is used in this situation.

   - Example: Continue with Step 17. WHEN S/G level greater than 17%, THEN do steps 16c through 16g.

REV. 01000 Controlled Copy #______

ALARM RESPONSE PROCEDURE **AR-C-10**

ALARM TITLE:
CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM

ALARM SETPOINT (S) :
CONTAINMENT RECIRC FAN SW OUT VALVE (FCV-4561) FULL OPEN
OR
CONTAINMENT RECIRC FAN SW OUT BYP VALVE (FCV-4562) FULL OPEN
AND
SW FLOW FROM ANY CONTAINMENT RECIRC FAN LESS THAN 1050 GPM

SOURCE (S) :
CNMT RECIRC FAN SW Outlet Flow
FIA 2033, FIA 2034, FIA 2035, FIA 2036

FCV-4561 or 4562 Full Open
LS-1 33/4561
LS-1 33/4562

REQUIRED ACTION:

1. Verify at least one of the following status lights bright:
   - RECIRC FN SW OUT FCV-4561 OPEN
   - RECIRC FN SW BYP FCV-4562 OPEN

2. Verify at least two Service Water Pumps operating
3. **<u>IF</u> alarm is due to loss of SW pump(s), <u>THEN</u> refer to AP-SW.2.**

4. Notify AO to perform the following: (inside Door #37)
   - Check CNMT Coolers SW outlet FCV-4561
   - Check CNMT Fan Coolers SW outlet bypass FCV-4562
   - Check CNMT Recirc Fans Coolers outlet flows
   - Check CNMT Recirc Fans Coolers outlet temperature
   - Report back on equipment status

5. <u>IF</u> SW leak indicated, <u>THEN</u> refer to AP-SW.1.

COMMENTS:
<u>References</u>:
- STATION SERVICE COOLING WATER SAFETY RELATED (SW) P&ID #33013-1250 SHEET 3
- ELEMENTARY #10905-369

- FCV-4561 is automatically positioned to control Containment Temperature.
- FCV-4561 fails open on Train A SI. FCV-4562 fails open on Train B SI.
- PPCS provides indication by digital points F2033D, F2034D, F2035D, F2036D which CRFC has low flow.

Continued on the next Page

Applicable To:

R. E. Ginna Nuclear Station

Approval Authority: Manager - Operations

A. PURPOSE - This procedure provides the necessary instructions to respond to a service water system leak.

B. ENTRY CONDITIONS/SYMPTOMS

1. ENTRY CONDITIONS - This procedure is entered from:

   a. ER-SH.1, RESPONSE TO LOSS OF SCREENHOUSE, if a SW leak has occurred.

2. SYMPTOMS - The symptoms of SERVICE WATER LEAK are:

   a. AR-PPCS-P2160, SERVICE WATER PUMPS A & B HEADER, or

   b. AR-PPCS-P2161, SERVICE WATER PUMPS C & D HEADER, or

   c. Sump pump down frequency rises in containment, the AUX BLDG, or INT BLDG, or

   d. Unexplained rise in the waste hold-up tank, or

   e. Visual observation of a SW leak, or

   f. Annunciator C-2, CONTAINMENT RECIRC CLRS WATER OUTLET HI TEMP 217°F, lit, or

   g. Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM, lit, or

   h. Annunciator E-31, CONTAINMENT RECIRC FAN CONDENSATE HI-HI LEVEL alarm, exhibits an unexplained rise in frequency, or

   i. Annunciator H-6, CCW SERVICE WATER LO FLOW 1000 GPM, lit.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify 480V AC Emergency Busses 17 and 18 - ENERGIZED**	Ensure associated D/G(s) running and attempt to manually load busses 17 and/or 18 onto the D/G(s) if necessary. IF neither bus 17 nor bus 18 can be energized, THEN perform the following: a. Trip the reactor b. WHEN all E-0 Immediate Actions done, THEN trip both RCPs c. Close letdown isol, AOV-427 d. Close excess letdown, HCV-123 e. Go to E-0, REACTOR TRIP OR SAFETY INJECTION IF either bus 17 OR bus 18 is deenergized, THEN refer to AP-ELEC.17/18, LOSS OF SAFEGUARDS BUS 17/18.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Verify At Least One SW Pump Running In Each Loop:** • A or B pump in loop A • C or D pump in loop B	Perform the following: a. Manually start SW pumps as necessary (257 kw each). b. IF adequate cooling can NOT be supplied to a running D/G, THEN perform the following: 1) Pull stop affected D/G 2) Immediately depress voltage shutdown pushbutton 3) Refer to ER-D/G.2, ALTERNATE COOLING FOR EMERGENCY D/Gs c. IF no SW pumps can be operated, THEN perform the following: 1) Trip the reactor 2) WHEN all E-0 Immediate Actions done, THEN trip BOTH RCPs 3) Close letdown isol, AOV-427 4) Close excess letdown, HCV-123. 5) Go to E-0, REACTOR TRIP OR SAFETY INJECTION d. IF only one SW pump can be operated, THEN refer to AP-SW.2, LOSS OF SERVICE WATER.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- o Abnormally low pressure in either SW loop may indicate that the idle pump check valve is open. This may be corrected by restarting or isolating the idle pump.
- o Low Pressure in either SW Loop may be a result of the running pump configuration.

3 Check SW System Status:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check SW loop header pressures: o Pressure in both loops - APPROXIMATELY EQUAL o PPCS SW low pressure alarm status - NOT LOW • PPCS point ID P2160 • PPCS point ID P2161 o Pressure in both loops - STABLE OR RISING	a. IF three SW pumps operating and either loop pressure less than 40 psig, THEN trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION. IF only two SW pumps operating and either loop pressure less than 45 psig, THEN start one additional SW pump (257 kw each pump).
b. Check SW loop header pressures - GREATER THAN 55 PSIG	b. IF either SW loop pressure is less than 55 PSIG with three SW pumps running AND cause can NOT be corrected, THEN initiate a controlled shutdown while continuing with this procedure (Refer to AP-TURB.5, RAPID LOAD REDUCTION).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: o If SW is lost to any safeguards equipment, the affected component should be declared inoperable and appropriate actions taken as required by ITS, Section 3.

o CNMT sump A level of 10 feet is approximately 6 feet 6 inches below the bottom of the reactor vessel.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Check For SW Leakage In CNMT:	
	a. Check Sump A indication o Sump A level - RISING -OR- o Sump A pump start frequency - RISING (Refer to RCS Daily Leakage Log)	a. IF the SW leak is NOT in the CNMT, THEN go to Step 6.
	b. Evaluate Sump A conditions: 1) Verify Leakage within capacity of one Sump A pump (50 gpm) 2) Check Sump A level - LESS THAN 10 FEET	b. Plant shutdown should be considered, consult plant staff.
	c. Direct RP to establish conditions for CNMT entry	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- o One Reactor Compartment cooling fan should be running whenever RCS temperature is greater than 135°F.
- o CNMT recirc fan condensate collector level indicators may be helpful in identifying a leaking fan cooler.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Check CNMT fan indications: o CNMT recirc fan collector dump frequency - NORMAL (Refer to RCS Daily Leakage Log) o CNMT recirc fan SW flows - APPROXIMATELY EQUAL (INTER BLDG basement by IBELIP) • Recirc Fan A, FIA-2033 • Recirc Fan B, FIA-2034 • Recirc Fan C, FIA-2035 • Recirc Fan D, FIA-2036 o Reactor compartment cooler SW outlet pressures - APPROXIMATELY EQUAL (INTER BLDG SAMPLE HOOD AREA) • Cooler A - PI 2232 • Cooler B - PI 2141	Dispatch AO with locked valve key to perform ATT-2.3, ATTACHMENT SW LOADS IN CNMT to determine leak location. <u>WHEN</u> CNMT SW leak location identified, <u>THEN</u> go to Step 9.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	Dispatch AO To Screenhouse To Perform The Following:	
	a. Verify idle SW pump check valve closed o Idle pump shaft stopped o Idle pump discharge pressure - ZERO (unisolate and check local pressure indicator) o SW Pump A, PI-2098, V-4501D o SW Pump B, PI-2099, V-4502D o SW Pump C, PI-2100, V-4503C o SW Pump D, PI-2101, V-4504C	a. Notify Control Room of any indication of check valve failure.
	b. Investigate for SW leak in Screenhouse - NO EXCESSIVE LEAKAGE INDICATED	b. Perform the following: 1) Identify leak location. IF excessive leakage from underground header indicated, THEN isolation of header should be considered (Refer to ATT-2.2, ATTACHMENT SW ISOLATION) 2) Notify Control Room of leak location.

STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

<u>NOTE</u>: Refer to ATT-2.2, ATTACHMENT SW ISOLATION for a list of the major non-safeguards loads supplied by each service water header.

7 Check Indications For Leak Location:

- o AUX BLDG sump pump start frequency - NORMAL (Refer to RCS Daily Leakage Log)
- o Annunciator L-9, AUX BLDG SUMP HI LEVEL - EXTINGUISHED
- o Annunciator L-17, INTER BLDG SUMP HI LEVEL - EXTINGUISHED

RESPONSE NOT OBTAINED:

Dispatch AO to the specific area to investigate for leakage.

<u>IF</u> leakage is from the common SW discharge header from the CCW and SFP Heat Exchangers, <u>THEN</u> perform the following:

a. Evaluate Leak Rate. If the leakage threatens to flood safeguards pumps (RHR or SI) <u>THEN</u> perform the following:

1) Trip the Reactor and perform E-0, REACTOR TRIP OR SAFETY INJECTION Immediate Actions

2) Trip both RCP's

3) Close AOV-427 and HCV-123

4) Close all Aux Building SW Isolation Valves

- o MOV-4616,4735
- o MOV-4615,4734

b. Place the SW Redundant Return Line in service per T-36.2, SERVICE WATER REDUNDANT RETURN LINE OPERATION.

c. Close SFP Heat Exchanger B SW outlet valve V-8685.

d. If the Aux Building SW isolation valve were closed <u>THEN</u> reopen the valves.

- o MOV-4616,4735
- o MOV-4615,4734

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	Dispatch AO To Locally Investigate For SW Leakage And To Monitor Operating Equipment • Turbine BLDG • SAFW pump room	

<u>NOTE</u>: If SW is lost to either D/G, refer to ER-D/G.2, ALTERNATE COOLING FOR EMERGENCY D/Gs, if cooling is required.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	Evaluate SW Leak Concerns	
	a. Check SW pump status - AT LEAST THREE PUMPS RUNNING	a. <u>IF</u> either SW header pressure less than 45 psig, <u>THEN</u> start third SW pump.
	b. Check SW loop header pressure - BOTH LOOPS GREATER THAN 45 PSIG	b. Perform the following: 1) Dispatch AO to split A and B SW headers (refer to ATT-2.5, ATTACHMENT SPLIT SW HEADERS) 2) <u>IF</u> plant at power, <u>THEN</u> initiate a controlled shutdown (Refer to AP-TURB.5, RAPID LOAD REDUCTION). 3) Go to Step 10.
	c. Verify leak location - IDENTIFIED	c. Return to Step 3.
	d. Verify plant operating at power	d. Verify SW system conditions appropriate for plant mode (Refer to ITS Section 3.7.8) and go to Step 10.
	e. Leak isolation at power - ACCEPTABLE	e. <u>IF</u> plant shutdown required, <u>THEN</u> refer to O-2.1, NORMAL SHUTDOWN TO HOT SHUTDOWN or AP-TURB.5, RAPID LOAD REDUCTION.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Dispatch AO(s) To Locally Isolate SW Leak As Necessary	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	Verify SW Leak Isolated	
	a. Monitor SW System Operation o SW loop header pressure - RESTORED TO PRE-EVENT VALUE Archive PPCS point ID loop A P2160 OR loop B P2161) o Both SW loop header pressures - STABLE	a. <u>IF</u> SW leak can <u>NOT</u> be isolated within the affected loop, <u>THEN</u> stop SW pumps in the affected loop.
	b. Verify At Least One SW Pump Running In Each Loop: • A or B pump in loop A • C or D pump in loop B	b. Perform the following: 1) Ensure two SW pumps running (257 kw each). 2) <u>IF</u> adequate cooling can <u>NOT</u> be supplied to a running D/G, <u>THEN</u> perform the following: a) Pull stop affected D/G b) Immediately depress voltage shutdown pushbutton c) Refer to ER-D/G.2, ALTERNATE COOLING FOR EMERGENCY D/Gs.

This Step continued on the next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 11 continued from previous page)

3) IF no SW pumps can be operated, THEN perform the following:

   a) Trip the reactor

   b) WHEN all E-0 Immediate Actions done, THEN trip BOTH RCPs

   c) Close letdown isol, AOV-427

   d) Close Excess Letdown Isolation Valve, HCV-123

   e) Go to E-0, REACTOR TRIP OR SAFETY INJECTION

4) IF only one SW pump can be operated, THEN refer to AP-SW.2, LOSS OF SERVICE WATER.

c. Restore to normal position all valves repositioned during leak investigation EXCEPT leak isolation boundary.

12 Evaluate MCB Annunciator Status (Refer to AR procedures)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Refer to 0-9.3, NRC IMMEDIATE NOTIFICATION, for reporting requirements.

13 Notify Higher Supervision

-END-

AP-SW.1 APPENDIX LIST

TITLE

1) ATTACHMENT SW ISOLATION (ATT-2.2)
2) ATTACHMENT SW LOADS IN CNMT (ATT-2.3)
3) ATTACHMENT SPLIT SW HEADERS (ATT-2.5)

Applicable To:

R. E. Ginna Nuclear Station

Approval Authority: Manager - Operations

CATEGORY 1.0

REVIEWED BY: ____________________

A. PURPOSE - This procedure provides the necessary instructions to respond to a loss of service water pumps.

B. ENTRY CONDITIONS/SYMPTOMS

1. ENTRY CONDITIONS - This procedure is entered from:

   a. AP-ELEC.17/18, LOSS OF SAFEGUARDS BUS 17/18.

   b. Any of several EOPs, when only one SW pump can be operated.

   c. ER-SH.1, RESPONSE TO LOSS OF SCREENHOUSE, when at least one SW pump is lost.

2. SYMPTOMS - The symptoms of LOSS OF SERVICE WATER PUMPS are:

   a. AR-PPCS-P2160, SERVICE WATER PUMPS A & B HEADER, or

   b. AR-PPCS-P2161, SERVICE WATER PUMPS C & D HEADER, or

   c. Annunciator C-2, CONTAINMENT RECIRC CLRS WATER OUTLET HI TEMP 217°F, lit, or

   d. Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050 GPM, lit, or

   e. Annunciator H-6, CCW SERVICE WATER LOW FLOW 1000 GPM, lit, or

   f. Annunciator H-9, AUXILIARY FEED PUMP CLG WTR FLTR HI DIFF PRESS, lit, or

   g. Annunciator I-10, CW PUMP SEAL WATER LO FLOW, lit, or

   h. Annunciator J-4, GENERATOR ISOPHASE BUS COOLING SYSTEM, lit, or

   i. Annunciator J-9, SAFEGUARD BREAKER TRIP, lit, or

   j. Annunciator K-30, TURBINE PLANT SAMPLING RACK TROUBLE, lit.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify 480V AC Emergency Busses 17 and 18 - ENERGIZED	Ensure associated D/G(s) running and attempt to manually load busses 17 and/or 18 onto their respective D/G(s). <u>IF</u> neither bus 17 nor bus 18 can be energized, <u>THEN</u> perform the following: a. Trip the reactor b. <u>WHEN</u> all E-0 Immediate Actions done, <u>THEN</u> trip both RCPs c. Close letdown isol, AOV-427 d. Close excess letdown, HCV-123 e. Go to E-0, REACTOR TRIP OR SAFETY INJECTION <u>IF</u> either bus 17 <u>OR</u> bus 18 is deenergized, <u>THEN</u> refer to AP-ELEC.17/18, LOSS OF SAFEGUARDS BUS 17/18.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Verify SW Pump Alignment:	
	a. Check at least one SW pump running in each loop • A or B pump in loop A • C or D pump in loop B	a. Perform the following: 1) Manually start SW pumps as necessary (257 kw each). 2) <u>IF</u> adequate cooling can <u>NOT</u> be supplied to a running D/G, <u>THEN</u> perform the following: a) Pull stop affected D/G b) Immediately depress voltage shutdown pushbutton 3) <u>IF</u> no SW pumps can be operated, <u>THEN</u> perform the following: a) Trip the reactor b) <u>WHEN</u> all E-0 Immediate Actions done, <u>THEN</u> trip BOTH RCPs c) Close letdown isol, AOV-427 d) Close excess letdown, HCV-123 e) Go to E-0, REACTOR TRIP OR SAFETY INJECTION 4) <u>IF</u> only one SW pump can be operated, <u>THEN</u> go to step 3. 5) <u>IF</u> at least two SW pumps can be operated, <u>THEN</u> go to step 8.
	b. Return to procedure or guidance in effect	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3 Align Alternate Cooling To One D/G (Refer to ER-D/G.2, ALTERNATE COOLING FOR EMERGENCY D/Gs):

- o IF A or C SW Pump is operating, THEN align alternate cooling to D/G B

-OR-

- o IF B or D SW Pump is operating, THEN align alternate cooling to D/G A

4 Isolate SW To Non-Essential Loads

a. Close screenhouse SW isolation valves

- MOV-4609
- MOV-4780

b. Close air conditioning SW isolation valves

- MOV-4663
- MOV-4733

c. Direct AO to perform Part C of ATT-2.2, ATTACHMENT SW ISOLATION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Monitor Plant Equipment Cooled By SW - TEMPERATURES STABLE • Exciter • MFP oil coolers • Instrument air compressors • Bus duct coolers • Seal Oil unit • Turbine lube oil cooler • CCW Hx • SFP Hx • AFPs • Condensate Pumps • Secondary sample coolers	IF required, THEN reduce load as necessary to stabilize equipment temperatures (Refer to O-5.1, LOAD REDUCTIONS, or AP-TURB.5, RAPID LOAD REDUCTION)
6	Notify Higher Supervision	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7 Check SW System Status:

a. Check SW loop header pressures:

- o PPCS SW low pressure alarm status - NOT LOW
  - PPCS point ID P2160
  - PPCS point ID P2161
- o Pressure in both loops - STABLE OR RISING
- o Check SW loop header pressures - GREATER THAN 40 PSIG

Response not obtained: a. Locally isolate selected SW loads as desired (Refer to ATT-2.2, ATTACHMENT SW ISOLATION)

b. Check at least one SW pump running in each loop:

- A or B pump in loop A
- C or D pump in loop B

Response not obtained: b. Perform the following:

1) Continue efforts to start at least one SW pump in each loop.
2) IF at least two SW pumps can be operated, THEN go to Step 8. IF NOT, THEN return to step 3.

8 Notify Higher Supervision

9 Select Operable SW Pumps For Auto Start

Response not obtained: Refer to ITS LCO 3.7.8

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Evaluate MCB Annunciator Status (Refer to AR Procedures)	
11	Return To Procedure or Guidance In Effect	

-END-

AP-SW.2 APPENDIX LIST

TITLE

1) ATTACHMENT SW ISOLATION (ATT-2.2)

Question #54 - Justification for Accepting Two Correct Answers Rev.1

Question #54 states the following:

"Given the following plant conditions:

- There is a tube rupture in the 'B' S/G
- The crew is performing actions to isolate the ruptured steam generator per E-3, STEAM GENERATOR TUBE RUPTURE
- 'A' S/G MSIV is closed

Which one of the following actions should be performed to stop/reduce the radioactive release in progress, per the Major Action Category isolation steps of E-3?"

The conditions given in the stem of the question place the crew at Step 4 in E-3. Step 4 in E-3 isolates flow from the ruptured S/G. Choices 'B', 'C', and 'D' address operation of the 'B' ARV and are plausible based on a review of E-3 Step 4.

Choice 'B' would be correct if the candidate interpreted from the stem that the action was being taken to minimize (reduce) the radioactive release associated only from an "uncomplicated" tube rupture in the 'B' S/G. This interpretation is based on the assumption that the 'B' ARV is operating properly in AUTO (E-3 Step 4.a). Note that, given the conditions stated and assuming an "uncomplicated" tube rupture, 'B' S/G pressure would be at ~1005 psig controlled by the steam dumps. 'B' S/G ARV would already be in AUTO at 1050 psig, and no adjustment as stated in choice 'B' would be required.

Choice 'D' would be correct if the candidate interpreted from the stem that a radioactive release WAS in progress. This is a reasonable interpretation based on the ambiguity of the words "... to stop/reduce the radioactive release in progress". In this case, the RNO action of Step 4.b would be required when the 'B' S/G pressure was less than 1050 psig. Given this interpretation and the fact that the 'B' S/G ARV would already be in AUTO at 1050 psig, it would be correct to conclude that 'B' S/G ARV is malfunctioning in AUTO, i.e. it is open at less than 1050 psig. In this case, the correct answer would be choice 'D' which places the 'B' ARV in manual at 0% demand (Closed) per the RNO of E-3 Step 4.b. Both B and D choices are reasonable actions which would be considered by a licensed RO/SRO in response to the given conditions. There is no information provided in the stem which would lead the candidate to believe that either of these actions would not be successful.

Therefore, choices 'B' and 'D' are both correct depending on a candidate's interpretation of the wording in the stem. Either interpretation is reasonable based on the ambiguity of the words "...stop/reduce the radioactive release in progress..."

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.11	
	Importance Rating	3.8		

Radiation Control - Ability to control radiation releases.

RO Question # 54 Rev 1

Given the following plant conditions:

- There is a tube rupture in the 'B' S/G
- The crew is performing actions to isolate the ruptured steam generator per E-3, STEAM GENERATOR TUBE RUPTURE
- 'A' S/G MSIV is closed

Which one of the following actions should be performed first to minimize a radioactive release, per the Major Action Category isolation steps of E-3?

A. Manually open the 'A' S/G ARV to maintain RCS temperature

B. Adjust 'B' S/G ARV controller to 1050 psig in auto

C. Shut the manual isolation valve for 'B' S/G ARV

D. Place the 'B' S/G ARV controller in manual at 0% demand

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the candidate might believe he should lower RCS temp (and ruptured S/G pressure) to prevent lifting a ruptured S/G ARV. This action is taken during the Cooldown phase, but is not used to control RCS temperature to prevent lifting the ruptured S/G ARV. With the intact S/G MSIV closed, steam dump is not available and the 'A' ARV should be set to maintain intact S/G pressure in AUTO.

B. Correct. The ruptured S/G ARV is adjusted to its normal setpressure to ensure that the ARV remains operable and opens BEFORE its associated first safety valve opens at 1085 psig.

C. Incorrect. Plausible because candidate might believe it was a conservative action to isolate a ruptured S/G ARV that was lifting normally in response to pressure. Nothing in stem states the ARV failed.

D. Incorrect. Same reasoning as 'C' – the candidate might believe he/she should take action to close a ruptured S/G ARV that was open.

Technical Reference(s): E-3 Background,
EOP Setpoint Document for H.3

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C 1.02 (As available)

Question Source: Bank # S019.0011

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
As Administered	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.11	
	Importance Rating	3.8		

Radiation Control - Ability to control radiation releases.

RO Question # 54

Given the following plant conditions:

- There is a tube rupture in the 'B' S/G
- The crew is performing actions to isolate the ruptured steam generator per E-3, STEAM GENERATOR TUBE RUPTURE
- 'A' S/G MSIV is closed

Which one of the following actions should be performed to stop/reduce the radioactive release in progress, per the Major Action Category isolation steps of E-3?

A. Manually open the 'A' S/G ARV to maintain RCS temperature

B. Adjust 'B' S/G ARV controller to 1050 psig in auto

C. Shut the manual isolation valve for 'B' S/G ARV

D. Place the 'B' S/G ARV controller in manual at 0% demand

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the candidate might believe he should lower RCS temp (and ruptured S/G pressure) to prevent lifting a ruptured S/G ARV. This action is taken during the Cooldown phase, but is not used to control RCS temperature to prevent lifting the ruptured S/G ARV. With the intact S/G MSIV closed, steam dump is not available and the 'A' ARV should be set to maintain intact S/G pressure in AUTO.

B. Correct. The ruptured S/G ARV is adjusted to its normal setpressure to ensure that the ARV remains operable and opens BEFORE its associated first safety valve opens at 1085 psig.

C. Incorrect. Plausible because candidate might believe it was a conservative action to isolate a ruptured S/G ARV that was lifting normally in response to pressure.

D. Incorrect. Same reasoning as 'C' – the candidate might believe he/she should take action to close a ruptured S/G ARV that was open.

Technical Reference(s): E-3 Background,
EOP Setpoint Document for H.3

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C 1.02 (As available)

Question Source: Bank # S019.0011

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION

o IF THE TDAFW PUMP IS THE ONLY AVAILABLE SOURCE OF FEED FLOW, STEAM SUPPLY TO THE TDAFW PUMP MUST BE MAINTAINED FROM ONE S/G.

o AT LEAST ONE S/G SHALL BE MAINTAINED AVAILABLE FOR RCS COOLDOWN.

* *

4 Isolate Flow From Ruptured S/G(s):

a. Adjust ruptured S/G ARV controller to 1050 psig in AUTO

b. Check ruptured S/G ARV - CLOSED

b. WHEN ruptured S/G pressure less than 1050 psig, THEN verify S/G ARV closed. IF NOT closed, THEN place controller in MANUAL and close S/G ARV.

IF S/G ARV can NOT be closed, THEN dispatch AO to locally isolate.

c. Close ruptured S/G TDAFW pump steam supply valve and place in PULL STOP

- S/G A, MOV-3505A
- S/G B, MOV-3504A

c. Dispatch AO with locked valve key to locally isolate steam from ruptured S/G to TDAFW pump.

- S/G A, V-3505
- S/G B, V-3504

d. Verify ruptured S/G blowdown valve - CLOSED

- S/G A, AOV-5738
- S/G B, AOV-5737

d. Place S/G blowdown and sample valve isolation switch to CLOSE.

IF blowdown can NOT be isolated manually, THEN dispatch AO to locally isolate blowdown.

- S/G A, V-5701
- S/G B, V-5702

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Complete Ruptured S/G Isolation:	
	a. Close ruptured S/G MSIV - RUPTURED S/G MSIV CLOSED	a. Perform the following: 1) Close intact S/G MSIV. 2) Place intact S/G ARV controller at 1005 psig in AUTO. 3) Adjust condenser steam dump controller to 1050 psig in AUTO. 4) Place condenser steam dump mode selector switch to MANUAL. 5) Adjust reheat steam supply controller cam to close reheat steam supply valves. 6) Ensure turbine stop valves - CLOSED. 7) Dispatch AO to complete ruptured S/G isolation (Refer to ATT-16.0, ATTACHMENT RUPTURED S/G, parts A and B). 8) Go to step 6.
	b. Dispatch AO to complete ruptured S/G isolation (Refer to ATT-16.0, ATTACHMENT RUPTURED S/G part A)	

Question #55 - Justification for Accepting Two Correct Answers Rev.1

Choice 'A' and choice 'B' are both correct for question #55 because the stem of the question does not limit the candidate to only the initial procedure entered in the response to the stated conditions.

Conditions in question #55 indicate a failure of R-17 without RCS in-leakage to the CCW system. As a result, both choices 'A' and 'B' are plausible.

The second part of question #55 states the following:
"What procedure(s) would be entered in response to these indications?"

Choice 'B' is correct because the E-16 Alarm Response would be the initial procedure used by the crew to respond to the alarm.

Choice 'D' is correct because STP-O-17.2 would be entered to perform initial assessment of detector operability and to perform post maintenance operability testing following repairs to the radiation monitor. The distracter analysis even recognizes the fact that the STP "would eventually be addressed, but would not be the first procedure entered." The question does not ask what the first procedure entered would be; it simply asks "what procedure(s) would be entered..."

The purpose of STP-O-17.2 is as follows:

- To test operability of Process and Iodine Radiation Monitors by performing the following:
  - Verify monitor responds properly to installed check source
  - Ensure High Alarms and Warning Alarms are left at values specified in P-9, Radiation Monitoring System.
  - Perform functional test.

STP-O-17.2 would be used to troubleshoot the abnormal conditions described and to perform post maintenance operability testing once repairs are completed. STP-O-17.2 is listed as a performance reference in S-14, *Area and Process Radiation Monitoring System*, and S-14.10, *Operation of Process Radiation Monitors (R-15 through R-20B).*

In summary, the second part of question #55 does not limit the candidate to the initial procedure entered in the response to the stated indications. Therefore, both choices 'A' and 'B' are correct.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
Proposed Re-write	Group #	1		
	K/A #	073	A2.02	
	Importance Rating	2.7		

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

RO Question # 55 Rev 1

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

- RMS channel R-17, Component Cooling Water, drawer display initially read 2.1E03 cpm, then rose rapidly to >1E06, and now reads "EEEEE"
- R-17 drawer WARN and HIGH lights are lit
- 40 gpm letdown orifice valve AOV-200B is in service
- PCV-135, letdown pressure control valve, is 40% open
- Both RCP labyrinth seal D/Ps are 40"

Which of the following (1) indicates the reason for these indications and (2) what is the procedure first entered in response to these indications?

A. (1) Detector failure
(2) STP-O-17.2, RAD MONITORS R-11 thru R-18 SOURCE CHECK, ALARM SETPOINT VERIFICATION, AND FUNCTIONAL TEST

B. (1) Detector failure
(2) E-16, RMS PROCESS MONITOR HIGH ACTIVITY

C. (1) RCS in-leakage to CCW system
(2) E-16, RMS PROCESS MONITOR HIGH ACTIVITY

D. (1) RCS in-leakage to CCW system
(2) AP-CCW.1, Leakage Into the CCW Loop

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the given indications indicate a detector failure <u>high</u> which over-ranged the circuit and activated the WARN and HIGH range alarm circuits in the

drawer. Part 2 is plausible because going to the STP-O procedure for checking setpoints and functionality would *eventually* be addressed, but would not be the first procedure entered. Incorrect because the E-16 alarm which accompanies the HIGH alarm is the higher priority procedure which should be entered initially.

B. Correct. The given indications indicate a detector failure high which over-ranged the circuit and activated the WARN and HIGH range alarm circuits in the drawer. The HIGH alarm will close RCV-017, the CCW vent valve (but that information is not provided) and provide an input into the E-16 annunciator. The E-16 Alarm Response procedure will provide further guidance (e.g., verify that automatic actions have occurred).

C. Incorrect. Part 1 is plausible because Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system. Incorrect because the plant parameters provided in the initial conditions indicate that neither the NRHX or thermal barrier HX is leaking. Part 2 is the correct procedure to be entered initially. Incorrect because there is no other information in the stem which indicates that a valid leak into the CCW system is likely.

D. Incorrect. Part 1 is plausible because Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system. Incorrect because the plant parameters provided in the initial conditions indicate that neither the NRHX or thermal barrier HX is leaking. Part 2 is plausible because it's the procedure which E-16 will direct transition to, but given the lack of supporting plant information to confirm a leak into the CCW system, is not the correct procedure to be entered initially.

Technical Reference(s): E-16
STP-O-17.2

Proposed References to be provided to applicants during examination: None

Learning Objective: R3901C, 4.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

Examination Outline Cross-reference:	Level	RO		SRO
As Administered	Tier #	2		
	Group #	1		
	K/A #	073	A2.02	
	Importance Rating	2.7		

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

RO Question # 55

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

- RMS channel R-17, Component Cooling Water, drawer display initially read 2.1E03 cpm, then rose rapidly to >1E06, and now reads “EEEEE”
- R-17 drawer WARN and HIGH lights are lit
- 40 gpm letdown orifice valve AOV-200B is in service
- PCV-135, letdown pressure control valve, is 40% open
- Both RCP labyrinth seal D/Ps are 40”

Which of the following (1) indicates the reason for these indications and (2) what procedure(s) would be entered in response to these indications?

A. (1) Detector failure
(2) STP-O-17.2, RAD MONITORS R-11 thru R-18 SOURCE CHECK, ALARM SETPOINT VERIFICATION, AND FUNCTIONAL TEST

B. (1) Detector failure
(2) E-16, RMS PROCESS MONITOR HIGH ACTIVITY

C. (1) RCS in-leakage to CCW system
(2) E-16, RMS PROCESS MONITOR HIGH ACTIVITY

D. (1) RCS in-leakage to CCW system
(2) AP-CCW.1, Leakage Into the CCW Loop

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the given indications indicate a detector failure high which over-ranged the circuit and activated the WARN and HIGH range alarm circuits in the

drawer. Part 2 is plausible because going to the STP-O procedure for checking setpoints and functionality would *eventually* be addressed, but would not be the first procedure entered. Incorrect because the E-16 alarm which accompanies the HIGH alarm is the higher priority procedure which should be entered initially.

B. Correct. The given indications indicate a detector failure high which over-ranged the circuit and activated the WARN and HIGH range alarm circuits in the drawer. The HIGH alarm will close RCV-017, the CCW vent valve (but that information is not provided) and provide an input into the E-16 annunciator. The E-16 Alarm Response procedure will provide further guidance (e.g., verify that automatic actions have occurred).

C. Incorrect. Part 1 is plausible because Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system. Incorrect because the plant parameters provided in the initial conditions indicate that neither the NRHX or thermal barrier HX is leaking. Part 2 is the correct procedure to be entered initially. Incorrect because there is no other information in the stem which indicates that a valid leak into the CCW system is likely.

D. Incorrect. Part 1 is plausible because Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system. Incorrect because the plant parameters provided in the initial conditions indicate that neither the NRHX or thermal barrier HX is leaking. Part 2 is plausible because it's the procedure which E-16 will direct transition to, but given the lack of supporting plant information to confirm a leak into the CCW system, is not the correct procedure to be entered initially.

Technical Reference(s): E-16
STP-O-17.2

Proposed References to be provided to applicants during examination: None

Learning Objective: R3901C, 4.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	11
	55.43	

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

Comments:

REV. 11 CONTROLLED COPY # 4

ALARM RESPONSE PROCEDURE	AR-E-16

ALARM TITLE:
RMS PROCESS MONITOR HIGH ACTIVITY

ALARM SETPOINT (S) :
Refer to P-9, Radiation Monitoring System

SOURCE (S) :
R-10A through R-20B High Activity

REQUIRED ACTION:

1. Ensure automatic actions have occurred where applicable.
2. Notify Shift Supervisor, Health Physics and Auxiliary Operator to make appropriate investigation.
3. Refer to AR-RMS.11 through AR-RMS.20B and ER-RMS.1
4. Refer to EPIP 1-13 Local Radiation Emergency and/or EPIP 2-3 Major Radioactive Release to the Lake
5. Refer to EPIP 1.0 to review for event classification
6. Refer to O-9.3 if necessary
7. Refer to CH-RETS-RMS-INOP.
8. Refer to ITS LCO 3.3.5 and 3.4.15.
9. Refer to the ODCM.

COMMENTS:

References: ELEMENTARY #10905-384

RESPONSIBLE MANAGER

7-25-2001
EFFECTIVE DATE

REV. 5

CONTROLLED COPY # 4

ALARM RESPONSE PROCEDURE **AR-RMS-17**

LOCATION: CONTROL ROOM

ALARM TITLE:

R-17 COMPONENT COOLING

ALARM SETPOINT (S) :

REFER TO P-9

SOURCE (S) :

R-17 MONITOR

REQUIRED ACTION:

1. Verify RCV-017 closed
2. GO TO AP-CCW.1
3. Direct RP to perform CH-PRI-CCW-LEAK to determine CCW leakage.

COMMENTS:

REFERENCES: AUXILIARY COOLANT COMPONENT COOLING WATER (AC) P&ID #33013-1245
Computer Points R17, R17H

RESPONSIBLE MANAGER

8-17-99

EFFECTIVE DATE

R.E. Ginna Nuclear Power Plant
TECHNICAL PROCEDURE

STP-O-17.2

PROCESS RADIATION MONITORS R-11 THRU R-18, R-20 THRU R-22 AND IODINE MONITORS R-10A AND R-10B SOURCE CHECK, ALARM SETPOINT VERIFICATION, AND FUNCTIONAL TEST

Revision 00100

Safety Related

CONTINUOUS USE

Applicable To:

- R.E. Ginna Nuclear Power Plant

Approval Authority:
Manager-Operations

GINNA STATION
START:
DATE: ________________
TIME: ________________
COMPLETED:
DATE: ________________
TIME: ________________

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
00100	PCR-12-01523	• Deleted fifth bullet in Step 6.13.1.10, Plant modification now bypasses storm drain. ECP-10-000310

TABLE OF CONTENTS

SECTION TITLE PAGE

1.0 Purpose

1.1 To test operability of Process and Iodine Radiation Monitors by performing the following:

- Verify monitor responds properly to installed check source.
- Ensure High Alarms and Warning Alarms are left at values specified in P-9, Radiation Monitoring System.
- Perform Functional Test.

2.0 Applicability/Scope

2.1. Reason for performing Surveillance:

☐ Scheduled Surveillance

☐ Post-Maintenance Functionality Verification (Enter Work Order Number ____________)

☐ Plant Conditions requiring test (explain in remarks)

Remarks: ____

____

2.2. This test may be performed in any MODE.

2.3. Surveillance Requirements satisfied by this procedure:

2.3.1. Technical Specifications

- SR 3.4.15.1, Channel Check of containment atmosphere radioactivity monitors.
- SR 3.4.15.2, Operational Test of containment atmosphere radioactivity monitors.

2.3.2. Offsite Dose Calculation Manual (ODCM)

- Table 3.1-2, Radioactive Liquid Effluent Monitoring Surveillance Requirements
- Table 3.2-2, Radioactive Gaseous Effluent Monitoring Surveillance Requirements
- Table 3-4, Area Radiation Monitor Surveillance Requirements

2.4. Sections of this procedure may be used to perform individual component testing when required (for example, retesting and increased surveillance), and the remaining sections and associated attachments marked N/A.

2.5. **IF** this procedure is used for Post Maintenance Testing, **THEN** with SRO approval, only the pages used to perform the test are required to be attached. This **SHALL** include Section 5.0, Subsections in Section 6.0 used to test the applicable component(s), Section 7.0 and any attachment(s) used during testing. Steps in these included sections and attachment(s) that are **NOT** performed are to be marked N/A.

2.6. The following radiation monitors are tested in this procedure:

- R-10A, Containment Vent Iodine
- R-10B, Plant Vent Iodine
- R-11, Containment Vent Particulate
- R-12, Containment Vent Noble Gas
- R-13, Plant Vent Particulate
- R-14, Plant Vent Noble Gas
- R-15, Air Ejector & Gland Seal Exhaust
- R-16, Containment Fan Coolers
- R-17, Component Cooling Water
- R-18, Liquid Waste Disposal
- R-20A, Spent Fuel Pool HX A
- R-20B, Spent Fuel Pool HX B
- R-21, Retention Tank
- R-22, High Conductivity Waste Tank

2.7. Subsections of Section 6.0 that are **NOT** performed may be marked N/A.

3.0 References And Definitions

3.1. Developmental References

3.1.1. ODCM, Section 3.0 – Radioactive Effluent Monitoring Instrumentation

3.1.2. P-9, Radiation Monitoring System

3.1.3. Radiation Monitoring System Operating and Maintenance Manual (Tracerlab/Victoreen)

3.1.4. Technical Specifications:

- Section 3.4.15, RCS Leakage Detection Instrumentation
- Section 3.7.10, Auxiliary Building Ventilation System (ABVS)

3.2. Performance References

3.2.1. A-52.4, Control of Limiting Conditions for Operating Equipment

3.2.2. A-52.12, Nonfunctional Equipment Important to Safety

3.2.3. P-9, Radiation Monitoring System

3.2.4. S-12.4, RCS Leakage Surveillance Record Instructions

3.3. Definitions

3.3.1. HV – High Voltage

4.0 Precautions and Limitations

4.1. **IF** any step in this procedure cannot be completed as stated, the Shift Manager **OR** Control Room Supervisor **SHALL** be notified immediately.

4.2. The Shift Manager **SHALL** be notified immediately **IF** any Acceptance Criteria is **NOT** met, **OR IF** any malfunction **OR** abnormal conditions occur.

4.3. The following MCB annunciators could alarm during the performance of this procedure:

- A-25, CONTAINMENT VENTILATION ISOLATION
- AA-2, AVT WATER TREATMENT PANEL TROUBLE
- E-16, RMS PROCESS MONITOR HIGH ACTIVITY
- E-20, CNMT OR PLANT VENT RAD MON PUMP TRIP
- K-27, DRAINAGE SYSTEM PH PANEL
- L-1, AUX BLDG VENT SYSTEM CONTROL PANEL

4.4. Source Check values were determined as approximately two standard deviations less than the historical mean value as determined by the Radiochemist.

6.9. **R-17 COMPONENT COOLING Monitor**

6.9.1. **VERIFY** R-17 CONTAINMENT COOLING digital readout is ILLUMINATED. ________

6.9.2. **PERFORM** the following to determine High Voltage value:

1. **DEPRESS AND HOLD** HV pushbutton. ________

2. **WHEN** High Voltage Reading on digital display stabilizes,
THEN RELEASE HV pushbutton **AND RECORD** value on Table 17. ________

3. **RECORD** Tape Value posted on drawer on Table 17. ________

INDEPENDENT VERIFICATION

4. **USING** Tape Value recorded on Table 17,
CALCULATE High Voltage Low Limit **AND** High Limit as follows,
AND RECORD values on Table 17:

__________ VDC - 6 VDC = __________ VDC
(Tape Value) (Low Limit) ________

____
IV

__________ VDC + 6 VDC = __________ VDC
(Tape Value) (High Limit) ________

____
IV

TABLE 17

Tape Value (VDC)	Low Limit (VDC)	High Voltage Reading (VDC)	High Limit (VDC)

5. **IF** High Voltage Reading is outside the Low Limit **AND** High Limit values recorded on Table 17,
THEN PERFORM the following:
OTHERWISE, **MARK** this Step N/A.

a. **DECLARE** R-17 INOPERABLE. ________

b. **NOTIFY** Shift Manager. ________

6.9.2.5 (Continued)

c. **INITIATE** a Condition Report to document failure. ________

6.9.3. **PERFORM** the following to determine As-Found alarm values:

1. Momentarily **DEPRESS** HIGH pushbutton **AND** **RECORD** High Alarm As-Found value on ATTACHMENT 1, SETPOINT DATA SHEET. ________

2. Momentarily **DEPRESS** WARN pushbutton **AND** **RECORD** Warning Alarm As-Found value on ATTACHMENT 1, SETPOINT DATA SHEET. ________

6.9.4. **PERFORM** the following to test automatic functions:

1. **SET** Warning Alarm Setpoint below the meter reading using ATTACHMENT 2, MONITOR SETPOINT ADJUSTMENT. ________

2. **VERIFY** the following:

   - WARN alarm light is FLASHING ________
   - Bar graph is ILLUMINATED in amber ________

3. **SET** High Alarm Setpoint below the meter reading **BUT** above the Warning Alarm Setpoint using ATTACHMENT 2, MONITOR SETPOINT ADJUSTMENT. ________

4. **VERIFY** the following:

   - HIGH alarm light is FLASHING ________
   - Bar graph is ILLUMINATED in red ________
   - MCB annunciator E-16, RMS PROCESS MONITOR HIGH ACTIVITY, is ILLUMINATED ________
   - CCW SURGE TK VENT, RCV-017, is CLOSED ________

6.9.5. **PERFORM** the following to determine Source Check value:

1. **RECORD** Rate Meter Indication on Table 18. ________

2. **DEPRESS AND HOLD** Check Source pushbutton. ________

3. **RELEASE** Check Source pushbutton **AND** **RECORD** Rate Meter Source Check Indication on Table 18. ________

TABLE 18

Source Check Indication (CPM)	Rate Meter Indication (CPM)	Source Check Increase (CPM)

INDEPENDENT VERIFICATION

4. **USING** values recorded on Table 18, **CALCULATE** Rate Meter Source Check Increase as follows **AND** **RECORD** on Table 18:

_____________ - ___________ = Source Check Increase
(Source Check) (As-Found) ________

____
IV

5. **IF** Source Check Increase value recorded on Table 18 is **NOT** greater than or equal to 500 CPM, **THEN PERFORM** the following: **OTHERWISE**, **MARK** this Step N/A.

   a. **DECLARE** R-17 INOPERABLE. ________

   b. **NOTIFY** Shift Manager. ________

   c. **INITIATE** a Condition Report to document failure. ________

6.9.6. **PUSH** ALARM ACK pushbutton. ________

6.9.7. **VERIFY** the following are **NOT** flashing:

- HIGH alarm light ________
- WARN alarm light ________

6.9.8. **PERFORM** the following to set alarm setpoints:

1. **SET** High Alarm Setpoint to P-9 Setpoint value recorded on ATTACHMENT 1, SETPOINT DATA SHEET using ATTACHMENT 2, MONITOR SETPOINT ADJUSTMENT. ________

   a. **RECORD** High Alarm As-Left setpoint on ATTACHMENT 1, SETPOINT DATA SHEET. ________

2. **SET** Warning Alarm Setpoint to P-9 Setpoint value recorded on ATTACHMENT 1, SETPOINT DATA SHEET using ATTACHMENT 2, MONITOR SETPOINT ADJUSTMENT. ________

   a. **RECORD** Warning Alarm As-Left setpoint on ATTACHMENT 1, SETPOINT DATA SHEET. ________

6.9.9. **ENSURE** CCW SURGE TK VENT, RCV-017, OPENS. ________

INDEPENDENT VERIFICATION

6.9.10. **PERFORM** an Independent Verification of the following setpoints:

1. Momentarily **DEPRESS** HIGH pushbutton, **AND VERIFY** High Alarm Setpoint equals P-9 Setpoint value recorded on ATTACHMENT 1, SETPOINT DATA SHEET. ________ IV

2. Momentarily **DEPRESS** WARN pushbutton **AND VERIFY** Warning Alarm Setpoint equals P-9 Setpoint value recorded on ATTACHMENT 1, SETPOINT DATA SHEET. ________ IV

Question #100 - Justification for Accepting Two Correct Answers

Question #100 places the crew in E-3, STEAM GENERATOR TUBE RUPTURE, with a loss of offsite power. As a result, instrument air is not available. Also, based on stated conditions, adverse parameter values are not in effect. Given these conditions, only choices 'C' and 'D' are plausible.

Question #100 asks the candidate to identify how the subsequent RCS depressurization will be performed.

The only difference between 'C' and 'D' is the value of PRZR level at which the depressurization is terminated. Step 19.c of E-3 states that the depressurization is terminated when the following two conditions are met:

1. RCS pressure less than ruptured S/G pressure and,
2. PRZR level is greater than 10%.

Therefore, choice 'C' is correct. However, choice 'D' is also correct since the PRZR level stated (30%) is greater than 10%. Note that both 'C' and 'D' are only correct if the depressurization is terminated prior to PRZR level reaching 75%.

Examination Outline Cross-reference:	Level	RO	SRO
Proposed Re-write	Tier #		1
	Group #		1
	K/A #	065	AA2.07
	Importance Rating		3.2*

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Whether backup nitrogen supply is controlling valve position.

SRO Question # 100 Rev 1

Given the following:

- The team is responding to a SGTR.
- Upon transition from E-0, Reactor Trip or Safety Injection, to E-3, Steam Generator Tube Rupture, a loss of offsite power occurred.
- RCS cooldown has been completed with the following plant conditions:
    - Containment pressure: 1.2 psig
    - PRZR level: Below narrow range indication
    - Ruptured SG pressure: 1030 psig
    - RCS pressure: 1400 psig

Which one of the following identifies how the subsequent RCS depressurization will be performed and the minimum required PRZR level to terminate the depressurization?

A. Using instrument air, open one PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%

B. Using instrument air, open one PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 30%

C. Align nitrogen to one PORV per ATT-12.0, ATTACHMENT N2 PORVS, and open that PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%

D. Align nitrogen to one PORV per ATT-12.0, ATTACHMENT N2 PORVS, and open that PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 30%

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the second part (termination criteria) is correct, and the use of instrument air to the PORV is provided as an alternative before the option of

using nitrogen. Step 14 of E-3 restores instrument air to CNMT if adequate air compressors are running. Incorrect because with the loss of offsite power adequate air compressors will not be running, so instrument air will not be re-established to CNMT when the depressurization step is reached. Therefore, one PORV with nitrogen will be used.

B. Incorrect. Plausible because the use of instrument air to the PORV is provided as an alternative before the option of using nitrogen. Step 14 of E-3 restores instrument air to CNMT if adequate air compressors are running. Also 30% PRZR level would be the criteria if adverse CNMT conditions existed. Incorrect because with the loss of offsite power adequate air compressors will not be running, so instrument air will not be re-established to CNMT when the depressurization step is reached. Therefore, one PORV with nitrogen will be used. Also, adverse CNMT conditions do not exist, so the correct criteria for stopping the depressurization is 10% PRZR level.

C. Correct. With instrument air in CNMT unavailable, E-3 directs the alignment of nitrogen to the PORV per attachment-12.0. With normal CNMT conditions, the RCS will be depressurized until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%.

D. Incorrect. Plausible because the response is correct except for the PRZR level. Incorrect because with normal CNMT conditions the RCS will be depressurized until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%.

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

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: This is an SRO Only question because it requires knowledge of when to implement attachments, including how to coordinate them with procedure steps. The examinee requires SRO knowledge of the E-3 procedure to recognize that with the loss of offsite power, instrument air to CNMT will not be reset even though there is a step that would do this if adequate air compressors were available. This will use of nitrogen to the PORV. Additionally, the examinee must recognize that normal CNMT conditions exist, and that this results in 10% PRZR level criteria rather than the 30% criteria that would be used if adverse CNMT conditions existed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	065	AA2.07
	Importance Rating		3.2*

As Administered

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Whether backup nitrogen supply is controlling valve position.

SRO Question # 100

Given the following:

- The team is responding to a SGTR.
- Upon transition from E-0, Reactor Trip or Safety Injection, to E-3, Steam Generator Tube Rupture, a loss of offsite power occurred.
- RCS cooldown has been completed with the following plant conditions:
  - Containment pressure: 1.2 psig
  - PRZR level: Below narrow range indication
  - Ruptured SG pressure: 1030 psig
  - RCS pressure: 1400 psig

Which one of the following identifies how the subsequent RCS depressurization will be performed?

A. Using instrument air, open one PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%

B. Using instrument air, open one PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 30%

C. Align nitrogen to one PORV per ATT-12.0, ATTACHMENT N2 PORVS, and open that PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%

D. Align nitrogen to one PORV per ATT-12.0, ATTACHMENT N2 PORVS, and open that PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 30%

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the second part (termination criteria) is correct, and the use of instrument air to the PORV is provided as an alternative before the option of

using nitrogen. Step 14 of E-3 restores instrument air to CNMT if adequate air compressors are running. Incorrect because with the loss of offsite power adequate air compressors will not be running, so instrument air will not be re-established to CNMT when the depressurization step is reached. Therefore, one PORV with nitrogen will be used.

B. Incorrect. Plausible because the use of instrument air to the PORV is provided as an alternative before the option of using nitrogen. Step 14 of E-3 restores instrument air to CNMT if adequate air compressors are running. Also 30% PRZR level would be the criteria if adverse CNMT conditions existed. Incorrect because with the loss of offsite power adequate air compressors will not be running, so instrument air will not be re-established to CNMT when the depressurization step is reached. Therefore, one PORV with nitrogen will be used. Also, adverse CNMT conditions do not exist, so the correct criteria for stopping the depressurization is 10% PRZR level.

C. Correct. With instrument air in CNMT unavailable, E-3 directs the alignment of nitrogen to the PORV per attachment-12.0. With normal CNMT conditions, the RCS will be depressurized until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%.

D. Incorrect. Plausible because the response is correct except for the PRZR level. Incorrect because with normal CNMT conditions the RCS will be depressurized until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%.

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

Proposed References to be provided to applicants during examination: None

Learning Objective: REP03C 2.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: This is an SRO Only question because it requires knowledge of when to implement attachments, including how to coordinate them with procedure steps. The examinee requires SRO knowledge of the E-3 procedure to recognize that with the loss of offsite power, instrument air to CNMT will not be reset even though there is a step that would do this if adequate air compressors were available. This will use of nitrogen to the PORV. Additionally, the examinee must recognize that normal CNMT conditions exist, and that this results in 10% PRZR level criteria rather than the 30% criteria that would be used if adverse CNMT conditions existed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION

o THE PRT MAY RUPTURE IF A PRZR PORV IS USED TO DEPRESSURIZE THE RCS. THIS MAY RESULT IN ABNORMAL CNMT CONDITIONS.

o CYCLING OF THE PRZR PORV SHOULD BE MINIMIZED.

o THE UPPER HEAD REGION MAY VOID DURING RCS DEPRESSURIZATION IF RCPS ARE NOT RUNNING. THIS MAY RESULT IN RAPIDLY RISING PRZR LEVEL.

* *

NOTE: o If auxiliary spray is in use, spray flow may be enhanced by closing normal charging valve AOV-294 and normal PRZR spray valves.

o When using a PRZR PORV select one with an operable block valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	Depressurize RCS Using PRZR PORV To Minimize Break Flow And Refill PRZR:	
	a. Verify IA to CNMT - AVAILABLE	a. Refer to ATT-12.0, ATTACHMENT N2 PORVS to operate PORVs.
	b. PRZR PORVs - AT LEAST ONE AVAILABLE	b. IF auxiliary spray available, THEN return to Step 18b. IF auxiliary spray can NOT be established, THEN go to ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 1.

This Step continued on the next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

(Step 19 continued from previous page)

c. Open one PRZR PORV until ANY of the following conditions satisfied: o PRZR level - GREATER THAN 75% [65% adverse CNMT] -OR- o RCS pressure - LESS THAN SATURATION USING FIG-1.0, FIGURE MIN SUBCOOLING -OR- o BOTH of the following: 1) RCS pressure - LESS THAN RUPTURED S/G PRESSURE 2) PRZR level - GREATER THAN 10% [30% adverse CNMT]	c. IF auxiliary spray available, THEN return to step 18b. 1) IF auxiliary spray can NOT be established, THEN go to ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 1.
d. Close PRZR PORVs	d. IF either PRZR PORV can NOT be closed, THEN close associated block valve.

Q#4 - NRC Response to Post-Exam Comment

The NRC DOES NOT AGREE with the proposed change. RO Question #4 Key Answer Choice A is correct and there are NO other correct answers. This question will remain as-is on the exam.

Four of eleven applicants missed this question. All four selected Distracter Choice C. None of the applicants asked for any clarification of the question during exam administration.

RO Question #4 asks which one of the statements describes a basis, as explained in P-12, ELECTRICAL SYSTEMS PRECAUTIONS, LIMITATIONS, AND SETPOINTS, for why the generator trip circuit is designed to be time-delayed, such that the generator trip occurs later than the turbine trip on most turbine trips.

The licensee has proposed Distracter Choice C as a second correct answer. Choice C states "on a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds to prevent a power-to-flow concern upon reactor trip." This statement is a good distracter, in that it contains elements of a valid basis in P-12. The related basis in P-12 describes the increased severity consequence of a reactor trip on overpower, over-temperature, or low coolant flow if the generator trips immediately when the turbine trips and subsequently the electrical buses supporting RCP operation fail to transfer automatically to their off-site source. As described in P-12, an overpower, over-temperature or low-pressure trip, coincident with loss of forced circulation could make the consequences of the accident more severe than reported in the FSAR.

However, the statement in Distracter Choice C is not a correct answer to the question because it does not describe a basis for the time-delay. The distracter provides a reason for the protection provided, "to prevent a power-to-flow concern upon reactor trip," but ties that reason to an incorrect specific initiating condition, that of "a Turbine Trip causing a Reactor Trip." P-12 does not support the conclusion that the delay in removal of power from RCP motors is required for a turbine trip causing a reactor trip. In fact, a reactor trip caused by a turbine trip is intended to protect against an over-temperature condition.

UFSAR Section 7.2.2.2.13, which is referenced by the licensee in their recommendation for two correct answers, actually supports the NRC contention of only one correct answer. Section 7.2.2.2.13 describes the reactor trip on turbine trip as an anticipatory reactor trip that will work to avoid the resulting thermal transients that could otherwise result. If the main turbine tripped at a high reactor power and there was no reactor trip on turbine trip protective function then the reactor would subsequently trip on either over-temperature delta T or high pressure to prevent exceeding reactor safety limits. A turbine trip, per design, should not result in a challenge to the over-temperature delta T trip protection function because of the anticipatory reactor trip on turbine trip design protective function.

Distracter Choice C links some true statements, but makes an overall incorrect statement. The first part of the statement, "on a Turbine Trip causing a Reactor Trip the RCP is locked at 60 HZ for 60 seconds," is correct. The generator continues to supply power to RCPs for 60 seconds after **ANY** turbine trip, including those turbine trips at

high enough reactor power level such that the reactor trips on direct interlock when the turbine trips. The second part of the statement, "to prevent a power-to-flow concern upon reactor trip," describes why it is important (under certain conditions) to maintain power to the RCPs for a short period after a reactor trip. Joining the two parts, however, results in an incorrect statement because, per P-12, the prevention of a power-to-flow concern is based on overpower, over-temperature or low pressure reactor trips, not a reactor trip on a turbine trip.

Q#26 - NRC Response to Post-Exam Comment

The NRC DOES NOT AGREE with the proposed change. RO Question #26 Key Answer Choice B is correct and there are NO other correct answers. This question will remain as-is on the exam.

Five of eleven applicants missed this question. All five selected Distracter Choice D. None of the applicants asked for any clarification of the question during exam administration.

The licensee has proposed Distracter Choice D as a second correct answer. This question challenges the applicant to differentiate between Distracter Choice D and Key Answer B on the basis of identifying whether the SLB accident or the LOCA is the design basis accident for the peak pressure limit in Containment.

The question does not ask for all accidents analyzed in regard to containment design criteria. Rather, it asks for the design basis accident for the peak pressure limit in Containment. The basis for TS 3.6.4 explains that both of these accidents, LOCA and SLB, were analyzed as limiting design basis accidents to predict containment pressure transients. The analyses determined the SLB generates larger mass and energy releases than the worst-case LOCA and that the SLB bounds the LOCA event from the containment peak pressure standpoint. Therefore, Key Answer B is the correct answer. There is one, and only one, correct answer.

Q#31 - NRC Response to Post-Exam Comment

The NRC DOES NOT AGREE with the proposed change. RO Question #31 Key Answer Choice D is correct and there are NO other correct answers. This question will remain as-is on the exam.

Four of eleven applicants missed this question. Three applicants selected Distracter Choice B (a wrong setpoint but the correct procedure). Only one applicant selected Distracter Choice C (the wrong procedure). None of the applicants asked for any clarification of the question during exam administration.

RO Question #31 gives some abnormal plant conditions at full power (a containment recirc air cooler service water low flow alarm and only one service water pump running) and asks the applicant to identify the alarm setpoint and whether it would be appropriate, per the low cooler flow alarm response procedure, to refer to the service

water leak procedure or to the loss of service water procedure. The alarm response, AR-C-10, directs the operator to refer to the loss of service water procedure if the alarm is due to loss of service water pumps, a condition provided in the question stem. Therefore, Key Choice D is a correct answer. AR-C-10 directs operators to refer to the service water leak procedure if a service water leak is indicated. No information was provided to indicate a service water leak. Therefore, Distracter Choice C is NOT a second correct answer.

The post-exam comment provided a rules-of-usage basis for "referring" to either procedure as a prudent operator action. There would not be anything necessarily wrong with referring to both procedures during an actual event to verify that all useful actions are being taken. However, the question asks the procedure which should be referenced **per the alarm procedure**. With the alarm procedure as a constraint, as previously explained, only Key Answer Choice D is correct.

<u>Q#54 - NRC Response to Post-Exam Comment</u>

The NRC <u>DOES NOT AGREE</u> with the proposed change to accept TWO correct answers. RO Question #54 Key Answer Choice B is correct and there are NO other correct answers. This question will remain as-is on the exam.

Two of eleven applicants missed this question. One applicant selected Distracter Choice D. The other applicant selected Distracter Choice A. None of the applicants asked for any clarification of the question during exam administration.

RO Question #54 states a tube rupture is in progress and asks the applicant to select the action that should be performed to stop/reduce the radioactive release in progress per the major action category isolation steps of E-3. The applicable major action category given in the E-3 background document is stated as "Identify and Isolate Ruptured SG(s)." The question stem states the crew is performing actions to isolate per E-3. No other specific plant indications are provided upon which to base a decision to take actions other than the one directed in E-3 Step 4 in the action/expected response (AER) column. In the absence of information indicating the AER column action is or would not be successful, the only correct answer is the key answer, to adjust the ruptured SG ARV controller to 1050 psig in AUTO and to check the valve closed.

The licensee argues the question wording is ambiguous and that, since the ARV is already in auto at 1050 psig, a normal at-power alignment, the applicant could reasonably assume a malfunction of the auto circuit allowing the ARV to be open. Under this condition they reason it would be appropriate for a knowledgeable applicant/operator to take E-3 Step 4 Response Not Obtained (RNO) column actions to place the ARV in manual with 0% output signal (the second answer proposed by the licensee as correct). The facility's comment states there is no information provided which would lead the examinee to believe either the key answer or the proposed second correct answer would not be successful in isolating the SG and that therefore it would not be reasonable for an applicant to select the third option provided in the procedure step.

The NRC disagrees with this rationale. There is no more information to support an assumption of an auto failure that can be addressed through manual operation than there is for an assumption that the ARV is stuck open requiring manual isolation. The information in the question stem does not support assuming either of these RNO column actions is necessary.

The NRC does not agree with the facility argument. However, for discussion sake, if the facility argument was accepted, then the facility proposed resolution of accepting two answers would not be appropriate. The question would likely have to be deleted. If one was to assume the ARV is open although in AUTO at 1050 psig set point with pressure less than set point, there would be no reason to assume that taking the ARV controller to manual at 0% output would be successful in closing the valve. Assuming the ARV is open although demanded closed by the controller, it might be just as valid to manually isolate the ARV as described in Choice C. With three correct answers or multiple divergent answers, the question would be deleted.

Q#55 - NRC Response to Post-Exam Comment

The NRC AGREES with the proposed change. RO Question #55 has TWO correct answers. Key Answer B is correct. Distracter Choice A is also a correct answer. Distracter Choices C and D are NOT correct answers.

Three of eleven applicants missed this question. Two applicants selected Distracter Choice A (correct reason but the wrong procedure). One applicant selected Distracter Choice D (wrong reason and wrong procedure). None of the applicants asked for any clarification of the question during exam administration.

The licensee proposes Distracter Choice A as a second correct answer. Key Answer B is correct because it properly identifies the reason for the indications as a detector failure and identifies an expected procedure entered in response to indications of a detector failure. Distracter Choice B also correctly identifies the conditions as indicating a detector failure and identifies a different procedure that would be entered in response to the indications. The alarm procedure listed in the key answer would be entered because of alarm actuation on the conditions given. The surveillance procedure listed in the proposed second correct answer would be used to determine operability. There are two, and only two, correct answers. Choices B and A will be accepted as correct.

Q#100 - NRC Response to Post-Exam Comment

The NRC DOES NOT AGREE with the proposed change. SRO Question #100 Key Answer Choice C is correct and there are NO other correct answers. This question will remain as-is on the exam.

One of seven SRO applicants missed this question. That applicant selected Distracter Choice D. None of the applicants asked for any clarification of the question during exam administration.

The question asks the methodology directed by E-3, "Steam Generator Tube Rupture" to perform the depressurization. E-3 Step 19 directs opening one PORV until RCS pressure is less than ruptured SG pressure AND pressurizer level is greater than 10% (or 30% for adverse containment conditions). Containment conditions in the question stem are not adverse, so the appropriate depressurization stopping point is when level is greater than 10%. The key answer identifies that, in order to depressurize, the operator will "open the PORV until RCS pressure is less than ruptured SG pressure and PRZR level is greater than 10%," which mirrors the guidance in the EOP.

EOP Step 19 depressurizes to stop primary to secondary leakage. It provides 3 sets of conditions for terminating the depressurization. The first is an upper bound on pressurizer level of 75% (65% adverse) to avoid water solid conditions. The second is a lower bound on pressure as indicated by a loss of subcooling to limit void formation in the RCS. The third set of conditions, that set which is tested by the question, is the combination of RCS pressure less than SG pressure AND pressurizer level greater than 10% (30% adverse). This third set is intended as the point where primary to secondary leakage is stopped due to removal of driving head and where presssurizer level is low in its indicating band. The WOG EOP background documents identify the chosen level setpoint as "PRZR level **just in range**, including [tolerances], not to exceed 50%...to ensure margin to filling the pressurizer for RCS inventory control."

The license has proposed accepting Choice D as a second correct answer because the pressurizer level in the distracter is greater than the minimum level required. In contrast to the key answer, Distracter Choice D uses the same wording, but requires pressurizer level greater than 30%. Since the question is asking the depressurization method of the EOP, which is to depressurize until greater than 10% for current conditions, the method is not properly described as continuing to depressurize until level is greater than 30%. Therefore, Distracter Choice D is not a second correct answer.