Three Mile Island Nuclear Generating Station
Written NRC Exam Facility Comments
May 2005

Question ID Number: #007

Concern or Problem:
The answer is technically incorrect.

Recommended resolution:
Change the correct answer to "B".

Justification:
The energy release to the Reactor Coolant Drain Tank by a 15 second lift of the Power Operated Relief Valve (PORV) does not result in challenge to the relief or the rupture disc. This was reviewed and confirmed through the site engineering group and demonstrated on the replica simulator. The answer originally provided is technically incorrect.

Attached References:
Engineering Calc

Examination Outline Cross-Reference Tier # 2

Evolution/System 007 Pressurizer Relief/Quench Tank Group # 1

K/A # K3.01 Page # 3.5-2 RO/SRO Importance Rating 3.3 3.6

Measurement Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Initial plant condition:
- 100% power.

Event:
- Reactor tripped.
- PORV lifted for 15 seconds.

Based on these conditions, identify the ONE selection below that describes the sequence of events, following the PORV lift and the effect on containment.

A. (1) WDG-V-1 relief valve opens to the vent header, the vent header and waste gas compressors maintain pressure.
(2) No affect on containment.

B. (1) RCDT sparger and the volume of water in the RCDT keep the pressure from reaching either the relief or rupture disk setpoints.
(2) No affect on containment.

C. (1) WDG-V-1 relief valve opens to isolated header, rupture disk blows.
(2) Activity is detected on RM-A-2.

D. (1) WDG-V-1 relief valve opens to containment rupture disk blows,
(2) RB ES pressure setpoint is exceeded.

Technical Reference Lesson Plan 11.2.01.119, Waste Gas Disposal, PPT-38, Rev. 10.
302-694, Waste Gas System, Rev. 43.
OP-TM-220-000, Reactor Coolant System, Section 2.1.26, Page 9, Rev. 4.

Open Exam Reference None.

Learning Objective IV.B.09.01

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT AB is isolated by RTI, plausible under original design that would have closed WDG-V-2 at 2 psig.

B INCORRECT sparger and volume are based on a 12 second lift, plausible if design basis is not known.

C CORRECT , relief lift does no good as it is to the isolated header, rupture disk blows, RM-A-2 will detect the activity after a time delay. CM-V-1-4 do NOT close until 4# isolation.

D INCORRECT relief does not relieve to containment, 4 psig is not challenged. Plausible manual vents go to containment, length of time to reach / exceed 4 psig setpoint may not be known.

Comments Examinee is required to know:

(1) Sequential overpressure protection for RCDT.
(2) Rupture Disk discharge is to the RCDT room inside containment.
(3) Reactor Trip Isolation closes WDG-V-3 and WDG-V-4, isolating the RB LP Vent Header from the Aux. Bldg LP Vent Header.

Question:
Initial conditions - 100% power

Event:
- Rx Trip
- PORV lifted for 15 seconds.

Evaluation requested:

Determine if RCDT water volume is sufficient to prevent reaching RCDT relief valve or rupture disk setpoints.

Assume min level (74.6%), max temp (110F) for one case. ***[Smith, Matthew G.]*** (Assume Pressurizer pressure 2400 psig.)
Assume max level (80.8%), min realistic temp (80F) for a second case. ***[Smith, Matthew G.]*** (Assume pressurizer pressure 1900 psig.)

Answer/Basis:
No information is given concerning the initial conditions of the fluid in the RCDT (WDL-T-3) or RCS, therefore the following assumptions are made:

1. The fluid level in the RCDT is within the normal operating band of 74.6% to 80.8% (basis is OP-TM-LWDS-0105, revision 0),
2. the fluid temperature in WDT-T-3 is within the normal operating band of 105 –110 degrees F. (basis is 1101-1, revision 72, page 41),
3. the pressure in WDL-T-3 is at the nominal operating value of 0 PSIG (14.7 PSIA),
4. WDL-T-3 contains 42.5 gallons per percent from 0% to 100% (basis is OP-TM-LWDS-0105, revision 0),
5. WDL-T-3 contains a total volume of 771 cubic feet (basis is 1101-1, revision 72, page 41),
6. the setpoint for WDL-V-1 (WDL-T-3 relief valve) is 40 PSIG, with a minimum acceptable lift point of 38 PSIG (basis is IISCP),
7. RC-V-18 (Pressurizer vent) is normally closed, and is closed for this configuration (basis is 302-650),
8. T-cold is > 329 degrees F. (PORV setpoint is 2450 PSIG),
9. the pressurizer steam temperature is 666 degrees F. (saturated steam at 2464.7 PSIA)

The mass flow capacity of the PORV is given by the following formula (basis is system design basis document SDBD-T1-220, section 3.2.10):

W = 51.5 X Area X Pressure X 0.95
W = lb / hr of steam
Area = area orifice, in ^2 = 0.94 in^2
Pressure = inlet pressure, PSIA

Minimum quench volume in WDL-T-3 occurs when the tank is at a minimum level with a maximum temperature.

74.6% X 42.6 gallons per percent X 0.1336806 ft^3 per gallon = 424 ft^3

At the maximum operating temperature of 110 degrees F. and a pressure of 14.7 PSIA, the density of the fluid is 61.864 lb / ft^3, and the enthalpy is 78.02 BTU / LB. The resulting energy content of the fluid is :

424 ft^3 X 61.864 lb / ft^3 X 78.02 BTU / lb = 2,046,491 BTU

The resulting initial volume in WDL-T-3 is:

424 ft^3 X 61.864 lb / ft^3 = 26,230 LBM

To determine the mass flow that is introduced into WDL-T-3 during the 15 seconds when the PORV is open, the following calculation is used:

W = 51.5 X Area X Pressure X 0.95
W = lb / hr of steam
Area = area orifice, in ^2 = 0.94 in^2
Pressure = inlet pressure, PSIA = 2450 PSIG + 14.7 = 2464.7 PSIA

W = 51.5 X 0.94 X 2464.7 X 0.95 = 113,350 LBM / HR

W = 113,350 LBM / HR X 1 HR / 3600 SEC X 15 SEC = 472 LB

At the pressurizer operating conditions of 2450 PSIG (2464.7 PSIA) and an assumed steam temperature of 666 degrees F., the steam enthalpy is 1097 BTU / LB and the density is 7.45 LBM / ft^3. The resulting energy content of the fluid entering WDL-T-3 during the PORV operation is:

472 LB X 1097 BTU / lb = 517,784 BTU

The final mass within WDL-T-3 is:

26,230 LBM + 472 LBM = 26,702 LBM

The total energy content of this fluid is:

2,046,491 BTU + 517,784 BTU = 2,564,275 BTU

The average energy content of the fluid is:

2,564,275 BTU / 26,702 LBM = 96.0 BTU / LBM

Assuming, for the initial evaluation, that the tank pressure remains at 14.5 PSIA, the temperature of a fluid at 14.5 PSIA and an enthalpy of 96.0 BTU / LB is 128 degrees F. This is not a sufficient temperature to create an increase in pressure in WDL-T-3.

For the second part of the evaluation, maximum quench volume in WDL-T-3 occurs when the tank is at a maximum level with a minimum temperature.

80.8% X 42.6 gallons per percent X 0.1336806 ft^3 per gallon = 460 ft^3

At the minimum operating temperature of 105 degrees F. and a pressure of 14.7 PSIA, the density of the fluid is 61.933 lb / ft^3, and the enthalpy is 73.02 BTU / LB. The resulting energy content of the fluid is :

460 ft^3 X 61.933 lb / ft^3 X 73.02 BTU / lb = 2,080,280 BTU

The resulting initial volume in WDL-T-3 is:

460 ft^3 X 61.933 lb / ft^3 = 28,489 LBM

To determine the mass flow that is introduced into WDL-T-3 during the 15 seconds when the PORV is open, the following calculation is used (NOTE: an RCS pressure of 2450 PSIG is used rather than the suggested 1900 PSIG since the former presents a more conservative estimate of the final temperature / pressure in WDL-T-3):

W = 51.5 X Area X Pressure X 0.95
W = lb / hr of steam
Area = area orifice, in ^2 = 0.94 in^2
Pressure = inlet pressure, PSIA = 2450 PSIG + 14.7 = 2464.7 PSIA

W = 51.5 X 0.94 X 2464.7 X 0.95 = 113,350 LBM / HR

W = 113,350 LBM / HR X 1 HR / 3600 SEC X 15 SEC = 472 LB

At the pressurizer operating conditions of 2450 PSIG (2464.7 PSIA) and an assumed steam temperature of 666 degrees F., the steam enthalpy is 1097 BTU / LB and the density is 7.45 LBM / ft^3. The resulting energy content of the fluid entering WDL-T-3 during the PORV operation is:

472 LB X 1097 BTU / lb = 517,784 BTU

The final mass within WDL-T-3 is:

28,489 LBM + 472 LBM = 28,961 LBM

The total energy content of this fluid is:

2,080,280 BTU + 517,784 BTU = 2,598,064 BTU

The average energy content of the fluid is:

2,598,064 BTU / 28,961 LBM = 89.7 BTU / LBM

Assuming, for the initial evaluation, that the tank pressure remains at 14.5 PSIA, the temperature of a fluid at 14.5 PSIA and an enthalpy of 89.7 BTU / LB is 122 degrees F. This is not a sufficient temperature to create an increase in pressure in WDL-T-3.

Summary:
Assuming nominal operating parameters within the RCDT and a PORV lift of 15 seconds at 2450 PSIG, the initial RCDT water volume is sufficient to prevent reaching RCDT relief valve minimum setpoint of 38 PSIG (52.7 PSIA).

Preparer: Mark Fauber 5/17/05

Peer Review: Michael Fitzwater 5/19/05

Manager Approval: Bradley Shumaker 5/19/05

Peer Review of Question:
Initial conditions - 100% power

Event:
- Reactor Trip
- PORV lifted for 15 seconds.

Determine if RCDT water volume is sufficient to prevent reaching RCDT relief valve or rupture disk setpoints.

Assume min level (74.6%), max temp (110F) for one case.
(Assume Pressurizer pressure 2400 psig.)
Assume max level (80.8%), min realistic temp (80F) for a second case.
(Assume pressure pressure 1900 psig.)

Review and conclusion:
Referenced the following:
1. SDBD- T1-232 Rev. 2 "SYSTEM DESIGN BASIS DOCUMENT for LIQUID RADIOACTIVE WASTE (WDL) SYSTEM (#232)"
2. VM-TM-0212 Dresser Pressurizer Code Safety Valves

The SDBD identifies the design basis of the RCDT relief quenching ability in section 3.2.1.

The maximum flow rate from all three valves is given as 760,000 lbm/hr (PORV and two Pressurizer Code Safety Valves) blowing for 14.4 seconds. This clearly bounds the test case of only the PORV relieving for 15 seconds in how much mass & energy is being delivered to the RCDT.

The quench capacity assumed in the SDBD is at a water temperature of 120°F which is greater than the 110°F of the test case. The quench mass of the test case is approximately 29,614 lbm which is less than the SDBD nominal mass of 34,700 lbm. Thus the quench capacity of the RCDT in the test case is greater than the assumed design in the SDBD.

The SDBD states that the maximum peak pressure expected is well below the relief valve and rupture disc setpoints.

Test case 1 & 2 has only the PORV relieving to the RCDT with each test case having a RCDT quench capacity greater than the SDBD design that assumes the PORV and two Pzr Code Safeties relieving. Test case 1 bounds case 2, and, it is concluded that the RCDT relief valve setpoint and rupture disc will not be challenged in test case 1 or 2.

I further reviewed the calculation provided by M. Fauber and agree with his conclusion.

M. D. Fitzwater 5/19/05 Peer Reviewer

coolant sources impermissible under the current discharge licenses. Discharge would then require dilution with significant amounts of untritiated water before discharge concentration limits could be met. Higher tritium concentrations in the plant would also increase the individual and total plant operational radiation exposure.

Accumulation of tritium within the cycle could conceivably also result in a total amount of tritium greater than could be released in a given period without exceeding the permitted calculated integrated offsite dose limit for that period.

The operating philosophy, but not the design, was therefore modified to avoid these problems by regularly discharging excess tritiated water. Reactor Coolant Evaporator distillate and Miscellaneous Waste Evaporator distillate have accordingly been discharged, rather than recycled, ever since startup (Refs. 263; 1023).

~~3~~3.2.1 ~~Pressurizer Relief Quench, Collection, Storage and Cooling~~

~~3~~3.2.1
Requirement 1: A Reactor Coolant Drain Tank (RCDT, "Quench Tank", WDL-T-0003) shall be provided within the Containment Building to suppress the steam relief from the pressurizer and receive water drained from the primary system.

The collection and quench functions are process requirements for normal operation only. The quench function in particular is not required to mitigate the consequences of any design basis loss of coolant accident or other design basis event inside containment. Maintenance of these functions is therefore not governed by technical specifications (Ref. 46).

Features:	Design Max. PORV/Relief Valve Flow Rate:	760,000 lbm./hr.
	- Maximum Duration:	14.4 sec.
	- Maximum Design Steam Quenched:	3030 lbm.
	- At:	580 psia saturated.
	Maximum Temperature Expected:	215°F
	Maximum Pressure Expected:	30 psig
	Nominal Quench Water Volume Required:	561 ft^3. 34,700 lbm.
	Quench Water Mass Range:	33,000 to 336,400 lbm.

- At Design Max. Quench Water Temperature: 120° F

Total Tank Volume: 780 ft^3.

Nitrogen Blanket Pressure: 16 ± 0.5 psia

(Refs. 1, pp. 26, 27; 332; 1020; 1051)

Basis: This function is required because the effluent from the PORV and code safety valves is potentially radioactive and must be contained within the WDL system. In case of loss of the fuel clad fission product barrier in a design basis event this steam is potentially highly radioactive and must be contained within the containment.

The suppression or "quench" function of the Reactor Coolant Drain Tank is a design feature provided to permit this discharge to be collected and contained in a reasonable volume without exceeding the required safety valve backpressure (see this Section, Requirements 2 and 3, below).

The RCDT also receives liquids and gasses vented from the Steam Generators and Control Rod Drive mechanisms (CRDM's) when these are vented for filling or draining of the primary system, from lantern rings of valves in the reactor coolant piping, and from leak off from the third seals of the Reactor Coolant Pumps. The RCDT provides these functions as a convenient collection point inside containment for forwarding to the Liquid Radioactive Waste System (Refs. 1, p. 26; 1020).

The expected blowdown from the first safety relief valve was 100,000 lbm./hr. The 760,000 lbm./hr design basis value was the expected saturated steam flow to the sparger at 580 psia (565 psig) if all three Pressurizer Safety Valves lift. The 580 psia steam condition is the expected pressure in the sparger header, based on this flow rate through the 56 sparger nozzles (Ref. 1051). A maximum peak pressure and temperature of 215°F and 30 psig could occur at the end of this transient (Ref. 1019). This is the design value. Babcock and Wilcox (B&W) calculated values of 209°F and 41 psia (26 psig) for the 3030 lbm. and 760,000 lbm./hr blowdown. These values are based on the sum of air and steam partial pressures, with steam at the vapor pressure for the calculated end-of-blowdown temperature, and air partial pressure determined by assuming the initial vapor volume as 100 % air, compressed to the void volume available at the end of blowdown and raised to the end-of-blowdown temperature (Ref. 1051).

The value for the initial water inventory used in this B&W analysis, 518.5 ft^3 (at 120°F), is slightly less than the minimum quantity stated above (33,000 lbm., or 537 ft^3 at 120°F), which is in the conservative direction.

The Reactor Coolant Drain Tank is designed to accommodate this design steam discharge from the Pressurizer within its design pressure and temperature (see Section 33.6.3.1, Requirement 6, below for design pressure and temperature).

The stated design values are all pertinent to the calculated performance of the quench function in the tank within its design pressure and temperature.

By April of 1968, refinements of transient studies showed a maximum total relief of 2650 lbm. of steam. The original design value of 3030 lbm. was however retained as a conservative value (Refs. 1055; 332).

By the fall of 1968 the maximum total value had increased to 3005 lbm. (for a rod withdrawal accident). In 1969, the maximum expected blowdown rate also increased to 781,000 lbm./hr., based on as-tested relief valve capacity (Refs. 1057; 1056). This required a re-evaluation of the Pressurizer relief valve discharge piping pressure drop, sparger inlet pressure, and relief valve backpressure (Ref. 1051). The change was design verified by V-1101-220-023, dated October 25, 1990 (Ref. 1198). The verification was primarily by Calculation C-1101-220-5360-035, dated October 19, 1990 (Ref. 1199).

Background: An early Babcock and Wilcox (B&W) analysis provided the quantity and initial temperature of water required (Cited in Refs. 339; 1018). Draft functional specifications (Refs. 1020; 978) revised the design basis initial temperature from 120°F to 110°F, but the original 120°F was finally used. The original blowdown was also revised from 600,000 lbm./hr. to 760,000 lbm./hr. (Refs. 332. 1019). See the discussion under Requirement 6, Background.

33.2.1
Requirement 2: The Reactor Coolant Drain Tank (WDL-T-0003) shall be furnished with an internal sparging header and nozzles and an internal recirculation spray nozzle (Refs. 1. pp. 25, 26; 1020).

33.2.1
Requirement 3: The sparging header and nozzle design shall, together with the relief valve discharge piping design, assure a relief valve back pressure of no more than 700 psig (Refs. 1051; 1055; 1057).

Features: "A total of 56 2-in. Schutte and Koerting Figure 314 steam nozzles of 304 stainless steel attach radially to the bottom half of [the 14-inch] vertical manifold in eight rows, each containing 7 nozzles, with the nozzle rows at 45 degree intervals around the manifold. A second nozzle through the top tank head, radially offset from the tank vertical centerline, attaches to a Spraying Systems Co. Spray Nozzle No. 1-1/2 H30630MC of 304 SS inside the tank" (Ref. 1, p. 26).

The 56 sparger nozzles are arranged so that their tips describe a cylinder about 48 inches in diameter, with a vertical spacing of 10-1/4 inches. The lowest is 5 inches above the tank shell to bottom head weld line. The recirculation nozzle has a conical spray pattern with a total included angle of 30° and is located midway between the tank ID and the sparger manifold OD (Ref. 1019).

Basis: The underwater sparger nozzle manifold is provided for dispersing and quenching the steam relieved by the PORV and Pressurizer relief valves, in the inventory of cold water which is always maintained within the tank (Refs. 1, p. 26; 1019; 1020; 1051; 1054).

The 700 psig limit on relief valve backpressure is the specified design value for the relief valve bellows (Ref. 1055).

The design of the pressurizer quench header and the type, model, submergence, location and number of its nozzles are all pertinent to the ability of the RCDT subsystem to meet (1) the calculated steam flow rate, mixing, and quench function performance (Ref. 1019), and (2) the maximum allowed backpressure for the PORV and code safety valves (Ref. 339; 332; 1051; 1055). The nozzle backpressure is the major part of the relief valve discharge backpressure (Ref. 1051).

The steam flow through the sparger nozzles induces water flow through the suction openings, resulting in intimate mixing of steam and water and prompt condensation. Schutte & Koerting suggested the following for this application:

1. Nozzle discharge a minimum of 2 ft. to 2.5 ft. from the tank wall,

2. Minimum submergence of the uppermost nozzle of 2 or 3 ft., and

3. Minimum vertical spacing for the 2" nozzle of 8 to 10 inches.

The sparging header and nozzle geometry were based on these recommendations (Ref. 1051; 1058), and on the individual nozzle performance (Ref. 1054).

The single recirculation spray nozzle is included to cool the tank steam space and to quench secondary steam generated within the tank, during post-quench recirculation cooling of the tank (Refs. 1, p. 26; 1020; 1019; 1051).

The size of the 10" tank nozzle to the sparging header is determined by the relief valve discharge piping size, which is in turn determined by the required pressure drop. The 14" internal header size may have been chosen for fabrication convenience, i.e., to give a large diameter to permit adequate space for welding on the 56 sparging nozzle nipples. No other basis for this size has been found or suggested (Ref. 1051).

33.2.1
Requirement 4: The design cooling water inventory and quench temperature shall be maintained in the Reactor Coolant Drain Tank (WDL-T-0003) during reactor operation.

Feature: Administrative controls are required to maintain the required water inventory and temperature (Ref. 1, p. 27).

Basis: A cooling water inventory is required to quench the steam released by the Pressurizer Relief Valves. The required design inventory and quench temperature and their basis are described above.

Manual operation and administrative controls are used to prevent automatic operation from draining the tank below permissible levels.

"In order for the Reactor Coolant Drain Tank to accomplish the above indicated discharges to it without jeopardizing its capability to suppress the design quantity of pressurizer relief, the water inventory in the tank is allowed to cycle between about 33,000 to 36,400 pounds. Administrative control of the pump out of this tank must be applied in order to maintain these limits of water inventory in it" (Ref. 1, p. 27).

33.2.1
Requirement 5: The Reactor Coolant Drain Tank (WDL-T-0003) shall be provided with overpressure protection.

Features: A rupture disk is provided in the manway on top of the RCDT, sized for 760,000 lbm./hr. of saturated steam at 70 psia (55 psig). With tolerances, the minimum relief pressure is 49.6 psig (Refs. 1019; 1051).

The tank is also relieved to the Waste Gas System by Waste Gas Relief Valve WDG-V-0001, rated at 372 scfm (for nitrogen) at 55 psig (Refs. 342; 980; 1, p. 28). The rupture disk, and not this valve, is relied on for code pressure protection (Ref. 1019). The valve ~~has been reset~~lift setpoint has been changed to ~~to~~ for 40 psig.

Basis: The Reactor Coolant Drain Tank design is governed by Section III-C of the ASME code. Overpressure protection is required by the code (Ref. 1178) and by good practice.

The rupture disk relieving capacity is based on the maximum total design relief capacity of all three reactor Pressurizer Safety Valves, as described above.

The relieving pressure is the tank design pressure (see Section 33.6.3.1, Requirement 6, below).

The Waste Gas Relief Valve capacity was simply the capacity of a commercially available relief valve in the desired two-inch size. This was judged to be adequate.

since the valve has neither a specific nuclear safety nor a code protection purpose (Ref. 342; 980). A concern was later raised whether this valve relief could overpressure the waste gas header and blow the liquid waste tank loop seals (Ref. 793), but this concern was resolved by demonstrating that, with the given valve capacity, relief of this valve "under the worst possible conditions" would not blow the loop seals (Refs. 757; 784).

The set pressure of the WDG-V-0001 Waste Gas Relief Valve was originally the same 55 psig as the RCDT rupture disk. The setpoint was subsequently changed to 40 psig in order to permit the valve to relieve at a pressure between the 30 psig maximum specified operating pressure and the 55 psig rupture disk value, so that the valve will relieve before the rupture disk.

This change was made to Lonergan drawing A-1760S by Drawing Change Notice DCN C 081301.

Adequate margin is provided for (1) accumulation (2) valve set pressure tolerance; and (3) uncertainties and tolerances in the 30 psig maximum specified operating tank pressure.

Equipment: WDL-T-0003 RCDT

"The reactor coolant drain tank is vertically leg mounted, 8.5 ft in diameter by 15 ft-2-3/4 in. over-heads, and has ASME F&D heads. It is constructed of 304 stainless steel with carbon steel I beam legs and is furnished by B&W."

"The reactor coolant drain tank is provided with liquid level, liquid and gas space temperature, and gas space pressure instrumentation. A manhole is provided for inspection and maintenance, and includes a rupture disc cover to provide over-pressure protection for the tank. A 10-in. diameter nozzle, through the top head of the tank on its vertical centerline, connects to a 10 ft - 6 in. long by 14 in. diameter vertical internal manifold" (Ref. 1, p. 25).

The sparger and spray nozzles are described above. All other required nozzles are conventional (Ref. 1, p. 26).

The gas space of this tank is nitrogen-blanketed as described in Section 33.9.8, below (Ref. 1, p. 26).

" . . . Except for the post-pressurizer relief cooling of its contents (which is automatic), ~~and closing of its normally open vent valve (which are automatic)~~; all functions of this tank are solely under administrative control" (Ref. 1, p. 26).

33.2.1
Requirement 6: The Reactor Coolant Drain Tank shall be provided with an external cooling system to remove heat after a design basis quench.

Features: The RCDT Heat Exchanger (WDL-C-0001) and Pump (WDL-P-0008) are provided for external cooling of the contents of the tank.

Required Heat Removal: 3.12×10^6 BTU

Required Heat Exchanger
Capacity: 19,000 B./hr

At Design RCDT Recirculation
Flow: 30 gpm

At Design RCDT Recirculation
Inlet: 160 °F

At Design Cooling Water
Inlet: 95 °F

Specified Heat Exchanger
Capacity: 200,000 B./hr

At Specified RCDT Recirculation
Flow:* 30 gpm

At Specified Cooling Water
Flow: 30 gpm.

At Specified RCDT Maximum Recirculation
Inlet:* 210 °F

At Specified RCDT Minimum Recirculation
Outlet:* 120 °F

At Specified Cooling Water
Inlet: 95 °F

Specified maximum time for return to
120°F: 16 hr

(Ref. 896)

The heat exchanger required average duty of 195,000 B./hr. just meets the Bill of Material requirement for reduction of the temperature to 120°F in 16 hours (Ref. 896, p. RH-64) when applied to removal of 3.12×10^6 BTU as required by the "Design

Criteria" (Ref. 1021). The 200,000 B./hr. average duty specified by the Bill of Material would result in a 15.6 hour cycle, if no heat exchanger fouling or other degradation has occurred.

* (Note that the duty is not determined by the product of the heat capacity, flowrate, and difference between maximum recirculation inlet and minimum recirculation outlet temperatures, since these temperatures occur at different times in the duty cycle.)

Basis: An external cooling system is required to remove heat from the RCDT at a rate sufficient to return the tank temperature to the design basis quench water temperature (120°F, above) within the specified 16 hours.

The 16-hour value has no absolute basis. It was chosen as a conservative value somewhat below the original Babcock and Wilcox (B&W) value of 18 hours, which was ". . .believed to be a reasonable time and would minimize the cooler size" (Ref. 1051).

The 30 gpm recirculation flow rate is based on the desire to provide the pump in a standard size (see below).

The cooling water temperature is the design basis maximum for intermediate closed cycle cooling water (see Section ~~3~~3.9.15, below).

The basis for the 30 gpm cooling water flow is undocumented, but appears to be an engineering choice based on matching cooling water and process flows to obtain similar temperature differences across both sides of the cooler, and a reasonable approach temperature and log mean delta T (LMDT). This assures a reasonable size, space requirement, and cost for the cooler.

Background: The original basis from B&W required cooldown to 120°F in 18 hours (Ref. 1051). Preliminary Gilbert Associates (GAI) design criteria specified cool down to 110°F in the shorter of (1) the expected period between pressurizer relief actuations, or (2) two hours, following a design basis pressurizer relief event (Ref. 1020).

No description of the basis for these shorter periods has been found. The 16-hour, 120 °F values were those finally specified. The Bill of Material says "The cooling cycle (210°F to 120°F) must be accomplished in 16 hours or less" (Ref. 896, p. RH-64).

The 110°F temperature is that specified by Babcock and Wilcox as the upper bound for the normal operating temperature ("105 -110 F." Ref. 646, Section 9.1.1, p. 119), but was not the 120°F design basis limit.

From December 1973 through August of 1974, excessive leakage from the Pressurizer

Relief Valves caused the temperature in the RCDT to rise above 200°F. The heat rate was above the capacity of the tank, pump, and heat exchanger to maintain the design 120°F initial quench temperature.

Various options were investigated to increase the cooling capacity of the external heat exchanger loop (Refs. 74; 72; 73; 71; 375; 358; 70; 69; 373; 374; 372; 370; 376; 371; 976; 356; 355; 349; 361; 366; 364; 361; 1168; 974; 362).

The need for additional cooling was subsequently eliminated by repair of the Pressurizer Safety Valves.

Change Mod No. 323 was issued to install an additional cooler, but was cancelled except for the installation of flanges in the cooling loop piping to accommodate the cooler installation (Ref. 425).

Equipment: WDL-C-0001 Reactor Coolant Drain Tank Heat Exchanger (RCDT HX)

33.2.1
Requirement 7: The vent of the WDL-T-0003 Reactor Coolant Drain Tank to the Reactor Building, through WDG-V-0134 and WDG-V-0135, shall be provided with means of preventing backflow of building atmosphere into the tank (Ref. 922).

33.2.1
Requirement 8: The design features necessary to meet Requirement 7, above, shall not compromise the design basis vent capacity of this line (Ref. 922).

Features: Requirements 7 and 8 are met by a "pop-off-plug" installed in the end of the vent line, consisting of an O-ring sealed plug designed to lift at 10 psig under high-flow conditions, and fitted with a small check valve to accommodate normal vent flows without allowing backflow (Refs. 922; 1170; 188).

The plug is classified "Important to Safety" (ITS) (Ref. 1169, p. 3)

Basis: The ". . . primary function of the plug is to prevent back-leakage of RB atmosphere into the RCDT . . .," which had contaminated the nitrogen blanket with oxygen. "The safety function of the plug, however, shall be to 'pop-off' clear of the vent line upon RCDT venting, at a pressure considerably less than 55 psig; the RCDT rupture disk burst pressure" (Refs. 922, p. 2; 1169, p. 3).

Background: In accident cases, large volumes of hydrogen and other non-condensible gasses may be relieved from the Pressurizer to the WDL-T-0003 Reactor Coolant Drain Tank (RCDT). This vent line is used for venting these gasses to the containment during post-accident recovery operations to prevent rupturing the RCDT rupture disc. This controlled release permits control of hydrogen concentration in the containment below

Question ID Number: #019

Concern or problem:

Question as stated gives no indication that the reliability of FW-P-1A has been assessed. Although this assessment is not a condition specified in Guide 15.1, CRS concurrence is a requirement. Some students, who were all tested at the CRS (SRO) level, did not want to give concurrence until that assessment was performed.

Recommended resolution:

Delete Question #19. A, C and D are all correct.

Justification:

CRS concurrence is a prerequisite of Guide 15.1. This is not given in the question stem, and could reasonably be withheld pending assessment of FW-P-1A reliability. As such, the option to leave EFW in service is a valid strategy, and the Emergency Operating Procedures do not establish a preference for MFW vs. EFW for the conditions listed.

Attached References:

Guide 15.1
FSAR 10.6.1 a.
E-mail dated 5/23/05 from Bill McSorely (EOP procedure writer.)

Question ID Number: #019

Concern or problem:

Question as stated gives no indication that the reliability of FW-P-1A has been assessed. Although this assessment is not a condition specified in Guide 15.1, CRS concurrence is a requirement. Some students, who were all tested at the CRS (SRO) level, did not want to give concurrence until that assessment was performed.

Recommended resolution:

Accept A or D as correct. (This does not change any grade, since no candidate chose D as a response.)

Justification:

Even in the event that the CRS withholds concurrence for returning EFW to standby, he should still use main feedwater preferentially. MFW source is from the condenser, and has very low oxygen. The water source for EFW is from the condensate storage tanks and is oxygen saturated.

Attached References:

Guide 15.1
FSAR 10.6.1 a.

Form ES-401-5 **Written Exam Question Worksheet** Q # 019

Examination Outline Cross-Reference Tier # 2

Evolution/System 061 Auxiliary/Emergency Feedwater (AFW) System Group # 1

K/A # A2.01 Page # 3.4-47 RO/SRO Importance Rating 2.5 2.6

Measurement Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Startup of MFW pump during AFW operation.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor operating at 60% power.
- FW-P-1A is the only available Main FW Pump.

Event:
- Reactor trip due to FW-P-1A trip.

Current plant conditions:
- All 3 Emergency Feedwater Pumps are operating.
  - EFW control valves EF-V-30A-D are controlling OTSG levels at setpoint.
- RCS subcooling margin is 41 degrees F and steady.
- FW-P-1A has been restarted, and is in HAND maintaining 0.1 mlbm/hr to each OTSG.

Based on these conditions identify the ONE selection below that describes required disposition of EFW equipment.

A. Return EFW systems to normal standby conditions.

B. Stop all EFW pumps, and manually close EF-V-30A-D.

C. Continue operating EFW as the preferred source of FW.

D. Continue operating EFW as a back up to the only operating MFW pump.

Technical Reference OP-TM-EOP-010, Guide 15.1 Return EFW to Standby, Page 31, Rev. 3.

Open Exam Reference None.

Learning Objective V.E.10.02

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. All of the conditions for returning EFW to standby are met, therefore EFW must be returned to standby. In accordance with OS-24, operators are responsible to recognize conditions which apply to EOP-010 Rules and Guides.

B INCORRECT because the prerequisites for EFW shutdown are not met.

Distracter is plausible because with the reactor shutdown and MFW available, there can be a misconception that EFW can be simply shutdown without returning the systems to standby.

C INCORRECT because the conditions for returning EFW to standby are met, therefore EFW must be returned to standby.

Distracter is plausible because EOP-010 Rule 4 identifies EFW as preferred source of FW under other operating conditions.

D INCORRECT because the conditions for returning EFW to standby are met, therefore EFW must be returned to standby.

Distracter is plausible because of perceived risk due to failure vulnerability with only one FW Pump operating.

Comments In this question the examinee is required to evaluate impact of Startup of one MFW pump during AFW operation; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations. The correct answer has the examinee apply procedure knowledge of Guide 15.1 AFW "return to standby criteria" to existing plant conditions, and recognize that AFW operation is no longer required. This same procedure, Guide 15.1, is used to return EFW to standby. Writer purposely did not identify Guide 15.1 title, Return EFW to Standby, for obvious reasons.

10.6 EMERGENCY FEEDWATER SYSTEM

10.6.1 DESIGN BASIS

a. Function

The Emergency Feedwater (EFW) system supplies feedwater to the Steam Generators, removing heat (including reactor coolant pump energy, decay and sensible heat) from the Reactor Coolant System to allow safe shutdown of the reactor. The system is not required for plant start-up, normal plant operations or normal shutdown. The system is used only during emergency conditions and periodic testing.

The EFW system can withstand a design basis event and a single active failure, while performing its function to allow safe shutdown of the reactor. A single active failure will not inadvertently initiate EFW, nor isolate the Main Feedwater systems. An exception to the single failure criteria is the loss of all A/C power (Section 14.1.2.8~~.c~~) event. In this event, the turbine driven pump alone will deliver the necessary EFW flow. Consideration of a single active failure within the EFW system or HSPS is not required due to the low probability of the event. The EFW system actuates on loss of both Main Feedwater pumps, low Steam Generator water level, loss of all four Reactor Coolant Pumps, or high Reactor Building pressure. The Heat Sink Protection System (HSPS), providing the actuation and OTSG water level control signals, is described in Section 7.1.4.

The EFW system will control feedwater flow to maintain water level in the Steam Generators. The water level setpoint is based on the status of the Reactor Coolant pumps. Steam Generator water levels are maintained higher when all Reactor Coolant pumps are off to promote natural circulation in the Reactor Coolant system. Level control for the EFW system is independent of the Integrated Control system (ICS).

b. Process Data

The EFW system delivers water to the Once Through Steam Generators (OTSG) from various water sources, pumps, valves and piping. Chapter 14 describes the design basis events for which EFW must function. The most demanding design basis event requiring EFW is a loss of normal feedwater (LOFW) with off-site power available (See Section 14.2.2.7). The LOFW event requires any two (2) of the three (3) EFW pumps to provide feedwater at 550 gallons per minute total to the OTSGs at 1050 psig for heat removal from the RCS. The minimum pump performance for the design basis LOFW event satisfies the flow rate requirements for all other events requiring EFW function.

Guide 15.1
Return EFW to Standby

When ALL of the following conditions are satisfied,

_____ SCM > 25°F
_____ Main Feedwater flow has been established to each available OTSG
_____ At least one reactor coolant pump is operating
_____ OTSG level > 20" in each available OTSG.
_____ RB pressure < 2 psig
_____ CRS concurrence has been obtained

then PERFORM the following to place EFW in standby.

1. **PLACE** the EFW control valves in Manual

_____ EF-V-30A _____ EF-V-30B
_____ EF-V-30D _____ EF-V-30C

_____ 2. **ENSURE** all EFW actuation switches (8) are in DEFEAT.

_____ 3. **CLOSE** EF-V-30A & D **and ENSURE** OTSG A level is maintained with Main FW

_____ 4. **CLOSE** EF-V-30B & C **and ENSURE** OTSG B level is maintained with Main FW

_____ 5. **PLACE** Train A **and** Train B EFW Actuation switches for Loss of RCPs **and** High RB Pressure in ENABLE. (4 switches)

_____ 6. **If** at least one FW pump is RESET, **then PLACE** Train A **and** Train B EFW Actuation for Loss of FWPs in ENABLE (2 switches)

_____ 7. **If** OTSG A level > 20" and OTSG B level > 20", **then PLACE** Train A **and** Train B EFW Actuation for Lo-Lo OTSG Level in ENABLE (2 switches)

_____ 8. **PLACE** EF-P-2A in Normal-after-stop

_____ 9. **PLACE** EF-P-2B in Normal-after-stop

_____ 10. **ENSURE** MS-V-10A is CLOSED **and CLOSE** MS-V-13A

_____ 11. **ENSURE** MS-V-10B is CLOSED **and CLOSE** MS-V-13B

12. **PLACE** each EFW control valve in AUTO **and SELECT** REMOTE setpoint

_____ EF-V-30A _____ EF-V-30B
_____ EF-V-30D _____ EF-V-30C

Smith, Matthew G.

F-om:	McSorley, William P
':	Monday, May 23, 2005 4:02 PM
	Smith, Matthew G.
Subject:	RE: Review of question #19

The question does not specifically address all of the requirements to place EFW in standby. (i.e. RB pressure, RCP operation & CRS concurrence).
To answer the question you must make assumptions about what is not said.
There is no evidence upon which to assume that an RCP is not operating or RB pressure > 2 psig.
It would be a reasonable assumption to either have or not have CRS concurrence.

Answer B is wrong.
Answer C or D are not well defined. There is no restriction on providing MFW to a OTSG with EFW actuated (if fact it is implied that this must be
done in order to satisfy the criteria to place EFW in standby). Depending on the rate of MFW supplied, either C or D would be correct.

Answers A, C or D should be accepted as correct

-----Original Message-----
From: Smith, Matthew G.
Sent: Monday, May 23, 2005 3:42 PM
To: McSorley, William P
Subject: Review of question #19

Action Required: Review Q#19 and respond.
Recommendation:

Bill,

Here is the text of question #19. Could you please respond and give your determination of which answer(s) is(are) correct?

Thanks,

Matt Smith

Initial plant conditions:
- **Reactor operating at 60% power.**
- **FW-P-1A is the only available Main FW Pump.**

Event:
- **Reactor trip due to FW-P-1A trip.**

Current plant conditions:
- **All 3 Emergency Feedwater Pumps are operating.**
  - **EFW control valves EF-V-30A-D are controlling OTSG levels at setpoint.**
- **RCS subcooling margin is 41 degrees F and steady.**
- **FW-P-1A has been restarted, and is in HAND maintaining 0.1 mlbm/hr to each OTSG.**

Based on these conditions identify the ONE selection below that describes required disposition of EFW equipment.

A. Return EFW systems to normal standby conditions.

B. Stop all EFW pumps, and manually close EF-V-30A-D.

C. Continue operating EFW as the preferred source of FW.

D. Continue operating EFW as a back up to the only operating MFW pump.

Guide 15.1
Return EFW to Standby

When ALL of the following conditions are satisfied,

- ______ SCM > 25°F
- ______ Main Feedwater flow has been established to each available OTSG
- ______ At least one reactor coolant pump is operating
- ______ OTSG level > 20" in each available OTSG.
- ______ RB pressure < 2 psig
- ______ CRS concurrence has been obtained

then PERFORM the following to place EFW in standby.

1. **PLACE** the EFW control valves in Manual

   ______ EF-V-30A ______ EF-V-30B
   ______ EF-V-30D ______ EF-V-30C

______ 2. **ENSURE** all EFW actuation switches (8) are in DEFEAT.

______ 3. **CLOSE** EF-V-30A & D **and ENSURE** OTSG A level is maintained with Main FW

______ 4. **CLOSE** EF-V-30B & C **and ENSURE** OTSG B level is maintained with Main FW

______ 5. **PLACE** Train A **and** Train B EFW Actuation switches for Loss of RCPs **and** High RB Pressure in ENABLE. (4 switches)

______ 6. **If** at least one FW pump is RESET, **then PLACE** Train A **and** Train B EFW Actuation for Loss of FWPs in ENABLE (2 switches)

______ 7. **If** OTSG A level > 20" and OTSG B level > 20", **then PLACE** Train A **and** Train B EFW Actuation for Lo-Lo OTSG Level in ENABLE (2 switches)

______ 8. **PLACE** EF-P-2A in Normal-after-stop

______ 9. **PLACE** EF-P-2B in Normal-after-stop

______ 10. **ENSURE** MS-V-10A is CLOSED **and CLOSE** MS-V-13A

______ 11. **ENSURE** MS-V-10B is CLOSED **and CLOSE** MS-V-13B

12. **PLACE** each EFW control valve in AUTO **and SELECT** REMOTE setpoint

   ______ EF-V-30A ______ EF-V-30B
   ______ EF-V-30D ______ EF-V-30C

Question ID Number: #024

Concern or Problem:

Correct answer is "A".
The question identifies a condition of a LOOP and ESAS actuation. No other specified failures have occurred. In this condition the emergency diesel is expected to come up to speed in 10 seconds. Reference FSAR 8.2.3.1.b. The SAR goes on to state that the time delay for a LOOP and simultaneous LOCA is 36 seconds considering signal generation, electrical start up, and injection pump start up and initiation.

The basis document of 1107-3 contains a note on starting air pressure. It states: "The diesel generator has the ability to start and load with an air pressure as low as 100#. Based on physical condition of the air start system, the diesel generator can be considered in reduced availability and may not meet the 10 second start/load criteria. Contact system engineering to address operability under degraded conditions when below 175 psig."

Recommended resolution:

Change answer key to A.

Justification:

The air start system is not indicated degraded in the question. With an intact air start system without degraded conditions the diesel generator has been demonstrated and tested to start and be ready to load in less than 10 seconds at less than 100 psig air start pressure. The procedure change that initiated this note identified and used this successful test of both emergency diesel generators at less than 100 psig as a basis for change. Therefore, given the conditions of this question the diesel is capable of meeting the design function. The answer originally provided is technically incorrect.

Attached References:

Procedure Change Safety Determination
FSAR 8.2.3.1.b
1107-3 Note
Engineering response to ILT exam issue

Form ES-401-5 **Written Exam Question Worksheet** Q # 024

Examination Outline Cross-Reference Tier # 2

Evolution/System 064 Emergency Diesel Generator (ED/G) System Group # 1

K/A # K6.07 Page # 3.6-9 RO/SRO Importance Rating 2.7 2.9

Measurement Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- NO Emergency Diesel Generators operating.

Event 1:
- Alarm A-1-2, DIESEL GEN 1A TROUBLE, actuation.

Field report:
- STARTING AIR PRESSURE LOW alarm is actuated on EG-Y-1A local panel.
- Air compressor has been isolated from BOTH starting air receivers in order to terminate the leak.
- Starting air pressure is now steady at 120 psig.

Event 2:
- Loss of offsite power (LOOP) concurrent with ES actuation.

Based on these conditions identify the ONE selection below that completes the following statement:
Emergency Diesel Generator EG-Y-1A will ________________.

A. start and meet all design basis requirements.

B. start but may NOT reach full speed to pick up electrical load within 10-second requirement.

C. attempt to start but trip when the Start Failure Relay (SFR) actuates.

D. NOT attempt to start because it is locked out by the Shutdown Relay (SDR).

Technical Reference 1107-3, Diesel Generator, Section 2.1.5 Air System - Limits/Precautions, Page 16, Rev. 110.

Open Exam Reference None.

Learning Objective IV.G.8.19

Question Source ✓ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Based on this low air pressure, EDG cranking RPMs are lower and it takes longer to startand accelerate to full speed. This lengthens how long it takes to begin electrical loading. The diesel generator must be capable of starting and loading within 10-seconds.

Distracter is plausible based on misconception that there is no impact on diesel starting until starting air pressure is less than 100 psig.

B CORRECT. The EDG is not operable with starting air pressure less than 175 psig. The diesel is capable of starting with air pressure as low as 100 psig, but depending how low air pressure is, may not be capable of meeting the 10-second start/load criteria. Final air pressure in the stem conditions was selected to be 120 psig to ensure the diesel could still start - but would take longer.

C INCORRECT because the start failure relay will not actuate.

Distracter is plausible because if the diesel did not start, then the start failure relay would actuate.

D INCORRECT. SDR does not actuate on any problem related to starting air pressure. The SDR will trip on a start failure, but will not prevent diesel start attempt on low air presssure.

Distracter is plausible because the SDR will shutdown the diesel and prevent it from restarting on a start failure.

Comments Modified per NRC request added the word "may" to distractor "B".

AR: 00335958-02

Engineering Responses to ILT Exam Issues:

Engineering will track and document responses to ILT Exam Issues with a PassPort action-tracking item. Each response will follow the format noted below. Currently there are two open questions. Additional questions, if any, will be provided to Whit Lopkoff who will generate the PassPort action-tracking item. Whit will consult with Jeff Goldman for assignment of this issue and a due date. Whit will also update the table in assignment #1 of this MREQ and distribute it twice per day with the latest status of open issues.

Response format:

Question: Verify minimum starting air pressure required to allow EDG to meet design basis.
Bkgd: I talked with Tom Flemming regarding the EDG question I posed to you this morning. His immediate answer was that he didn't know of any other operability guidance below that in the procedure (175 psig.) Could you please check with Bill McFarland to see if he is aware of any data which would support an EDG meeting design requirements at a lower pressure? Also, Tom mentioned that Dick Bensel may have some data to support a lower starting pressure. Tom will still investigate, but we would like this checked out as thoroughly as we can.

Answer/Basis:

Following our phone conversation earlier today, I did some research into the diesel procedures.

1107-3 contains the following note in section 2.1.5:

"The diesel generator has the ability to start and load with an air pressure as low as 100#. Based on the physical condition of the Air Start System, the affected diesel generator can be considered in reduced availability and may not meet 10 second start/load criteria. Contact system engineering to address operability under degraded conditions when below 175#."

The Alarm Response Procedure for Panel DGA/B provides the following guidance for "Starting Air Pressure Low" (DGA/B-3-1):

"MANUAL ACTION REQUIRED: If receiver air pressure is below 175# and EG-Y-1A/B is NOT running, then declare EG-Y-1A/B inoperable per Tech Spec requirements."

This guidance was added by PCR-00-0596. The Safety Determination for the PCR contains the following:
"Item 1: An Operability Limit for the Air Start System was not previously documented in any of the diesel generator controlling procedures; therefore, an Operability Limit is being provided by this PCR. Multiple PRG Meetings and diesel testing had been performed to determine and validate an Air Receiver Pressure Lower Operability Limit. Based on the testing and discussions, a value of 175# is selected as a conservative value. Testing has validated that the diesels will start with an Air Receiver pressure as low as 65#, but not within 10 seconds. The "A" Diesel started and was received <10 second "Ready-to-Load" status at 75#, the "B" Diesel was successfully tested at 95#. Without additional testing, a value of 175# will be a conservative operability limit. The value also provides Operations the means to receive the Alarm prior to requiring an operability determination. Based on the provision of information and previous discussions of the PRG, this change does not have the potential to adversely affect nuclear safety or safe plant operations."

I do not know when the testing was performed or where it is documented.

I hope this information is helpful.

For the conditions of a LOOP and ESAS actuation and an EDG starting air pressure of 120 psig the expected response of the EDG is to start and load within 10 seconds meeting the design requirements. This is based upon no other degraded conditions.

Rich Sievers, review by Valent,JR

f. Instrument Cable

1) In general, tray loadings do not exceed the appearance of 100 percent fill. In the few cases where trays appear to exceed 100 percent fill, calculations were done to verify less than 100 percent fill and therefore tray loading concerns are satisfied.

2) There are no other types of cables mixed in with instrumentation cabling.

8.2.3 SOURCES OF AUXILIARY POWER

8.2.3.1 Description Of Power Sources

Each auxiliary power source will have various degrees of redundancy and reliability as outlined below.

a. As described in Section 8.2.2.2, normal power supply to unit auxiliary loads will be provided through either one of the auxiliary transformers connected to the 230 kV substation buses. Power to these transformers can be provided from any one of four transmission circuits and the nuclear generating unit if operating.

b. Upon loss of the sources of power described in item a. above, power will be supplied from two automatic, fast-start diesel engine generators. These are sized so that either one can carry the required engineered safeguards load. The nameplate ratings of each emergency generator are: (1) 2750 kW at 0.8 power factor continuously with an expected availability of 95 percent providing there is an inspection every 24 months (with a 25% allowable grace period) in accordance with procedures prepared in conjunction with the applicable recommendations of the Fairbanks Morse Owners Group and those of the manufacturer for this class of stand-by service, (2) 3000 kW at 0.8 power factor for 2000 hours, and (3) 3300 kW at 0.8 power factor for not more than 30 minutes. The diesel engines are cooled by a jacket coolant system which transfers engine heat to a coolant liquid. The jacket coolant system is designed to dissipate excess heat from the engine and lube oil to the atmosphere through heat exchangers (radiators) which employ a fan driven directly from the engine. The jacket coolant temperature is maintained when the unit is not operating by a standby heater system. The function of the standby heater system is to maintain minimum jacket coolant temperature (120°F nominal) and lube oil temperature (90°F minimum). Coolant is circulated through a 24 kW standby electric heater, the lube oil heat exchanger, the water jacket, combustion air coolers, and the radiator fan gearbox oil cooler by the standby coolant pumps. An auxiliary electric heater maintains gearbox lube oil temperature at 45°F minimum. Operation of the diesel generator above 250 rpm automatically isolates the standby system, provided appropriate interlocks are satisfied.

When the unit is operating the jacket coolant temperature is controlled by a temperature control valve that directs water through the radiators or through a bypass line.

Each emergency generator will feed one of the 4160 V engineered safeguards buses. Each generator is capable of feeding the required safeguards loads of one 4160 V bus plus selected BOP manually applied emergency loads following any loss of coolant accident (LOCA). The diesel generator Engineered Safeguards block loading sequence is given on Table 8.2-11.

The diesel generator load tables, 8.2-8 and 8.2-9 show major loads typical of the heaviest loading on one D/G in the event the redundant diesel generator fails to start. The actual loading is tracked by C-1101-741-E510-005. See Reference 17. Diesel generator 1A is listed since this is the heaviest loaded diesel assuming that the 1B diesel is not available. In all cases the total load is less than the 2000 hr. rating of 3000 KW for the diesel generator.

Sufficient fuel is stored to allow one unit to supply post accident power requirements for 7 days based on the electrical loads shown on Tables 8.2-8, 8.2-9 and C-1101-741-E510-005. The LOOP/LOCA (Table 8.2-9) load is assumed to exist for the first 24 hours and then reduced loading as shown on Table 8.2-9 is assumed to continue for the next 6 days. Fuel supplied from the main storage tank is stored at each unit in a 550 gallon diesel generator day tank. Level switches automatically control the operation of an AC and redundant DC motor driven pump to maintain day tank fuel level. Additional level switches provide high and low level alarms.

The starting air system consists of a dual drive air compressor, two air reservoirs and controls located external to the engine designed to provide air at 225 to 250 psi. Starting air is directed through a manual shut-off valve and two air start solenoid operated valves and an air distributor system in the engine. A vent valve solenoid valve closes during the starting cycle. Two pressure switches indicate starting air being applied to the engine. A pressure gauge is mounted on the instrument panel and an alarm switch is provided to signal low starting air pressure.

The distributor includes one pilot air valve for each cylinder.

The air compressor is two stages with a loadless start feature. It is normally driven by an electric motor and can be, in an emergency, driven by a diesel engine by shifting belts from the motor to the engine. The engine is electric start and is provided with a separate 12 Vdc battery and charger.

The units are located in an annex on the opposite side of the building from the 230 kV substation and transformers and are separately enclosed to minimize the likelihood of mechanical, fire, or water damage.

Each diesel engine will be automatically started upon the occurrence of the following incidents:

1) Initiation of safety injection operation.

2) Overpressure in the Reactor Building.

3) Loss of voltage or degraded bus voltage detected by the undervoltage protection scheme on the 4160 V engineered safeguards bus with which the emergency generator is associated.

   For each Diesel Generator Automatic Start, automatic safety injection actuation and automatic overpressure in the reactor building actuation are sensed via the following relays: Two out of three 63Z2A/RC1, 63Z2A/RC2, 63Z2A/RC3 or two out of the three: 63Z1B/RC1, 63Z1B/RC2, 63Z1B/RC3. Manual actuations for safety injection or overpressure in the Reactor Building is sensed via 1X2A/RC or 1X1B/RC. Loss of voltage or degraded voltage is sensed by two out of three relays 27-1 through 27-6. Upon loss of the 4160 V bus voltage, the diesel generator unit will be automatically connected to its bus. The sequence to accomplish this following the starting signal will be as follows:

   Step 1 Automatic tripping of breakers on the bus.

   Step 2 After the unit comes up to speed and voltage, the emergency generator breaker will automatically close.

   Step 3 Automatic and manual starting of equipment as required for safe plant operation.

   Loss of voltage detection and diesel breaker automatic close signals both use two out of three logic.

   If there is a requirement for safeguards system operation coincident with the loss of voltage on the 4160 V bus, Step 2 will be followed by the automatic sequential starting of safeguards equipment.

   In the event one emergency generator does not come on the line when called for, the automatic starting sequence of components associated with this generator and bus will be blocked.

The automatic sequential loading of each diesel generator with safeguards auxiliaries will be accomplished in five blocks as described in Item c. of Section 7.1.3.2. These blocks have been selected so as to limit the maximum system voltage dip to approximately 30 percent. Safeguards control center starters have been specified to hold in at 10 percent below this value.

Starting of a diesel engine generator takes 10 seconds. For a simultaneous LOCA and loss of offsite power a delay time of 35 seconds is assumed in the safety analysis (Chapter 14, Reference 77) to allow for signal generation, electrical supply startup, injection pump startup and initiation of the pumped injection flows. The high pressure and low pressure injection systems are in the first loading block. See Table 8.2-11. If the system, rather than the emergency generators, continues to feed the safeguards buses at the time of a LOCA, safeguards loads will be started in the same five blocks in order to limit voltage dips. Safeguards loads which are running prior to the LOCA signal are not tripped and will continue to run. Therefore, core injection systems will be in operation in less than 25 seconds since diesel starting time would not then be a factor.

c. Should an engineering safeguard be followed by a loss of offsite power, time delay has been provided for the diesel generator breaker closure to assure that adequate time has elapsed since the opening of the bus feeder breakers to allow for voltage decay on the buses and for the shedding of other loads with under voltage relays.

8.2.3.2 Generator Breaker Closing Interlocks

a. The following conditions must be met in order to manually close the diesel breaker:

1. Synch. switch must be on
2. Breaker racked in
3. 81-59, 2 out of 3 matrix satisfied, diesel ready for loading
4. 86B, bus overload reset
5. 86G, Diesel Differential reset
6. Place generator breaker control switch to close when generator synchronizes with the bus

Safety Determination

Questions:

#3. Does this change have the potential to adversely affect nuclear safety or safe plant operations?

No, the PCR provides the following items for Safety Determination evaluation:
1) Provides Air Start System Operability Limit.
2) Provides Overspeed Trip Values for information.
3) Changed titles and typo's that do not affect the sequence or intent of the Alarm Response Procedure. The summary page only provides a quick reference to all of the alarms. The changes help to enhance the information of the procedure.

Item 1: An Operability Limit for the Air Start System was not previously documented in any of the diesel generator controlling procedures; therefore, an Operability Limit is being provided by this PCR. Multiple PRG Meetings and diesel testing had been performed to determine and validate an Air Receiver Pressure Lower Operability Limit. Based on the testing and discussions, a value of 175# is selected as a conservative value. Testing has validated that the diesels will start with an Air Receiver pressure as low as 65#, but not within 10 seconds. The "A" Diesel started and was received <10 second "Ready-to-Load" status at 75#, the "B" Diesel was successfully tested at 95#. Without additional testing, a value of 175# will be a conservative operability limit. The value also provides Operations the means to receive the Alarm prior to requiring an operability determination. Based on the provision of information and previous discussions of the PRG, this change does not have the potential to adversely affect nuclear safety or safe plant operations.

Item 2: During the performance of the Overspeed Test during the Spring 2000 Outage, it was noticed that the Acceptance Criteria was available in the Tech Manuals/SIL's (Service information Letters), but not in the Alarm Response Procedure. The allowable range for the Overspeed Trip was added to the Setpoints Section of the Procedure. The addition of pertinent information based on the Tech Manual/SIL's does not have the potential to adversely affect nuclear safety or safe plant operations.

Item 3: The corrections made do not affect the purpose or content of the procedure; therefore, does not have the potential to adversely affect nuclear safety or safe plant operations.

#4 Does this make changes in the facility as described in the safety analysis report?

No, the changes described do not make any changes to the facilities or equipment as described in the SAR. The changes provide more detailed information than previously contained in the procedures and do not affect the normal response of operations.

Question ID Number: #024

Concern or Problem:

Correct answer is "A".

The question identifies a condition of a LOOP and ESAS actuation. No other specified failures have occurred. In this condition the emergency diesel is expected to come up to speed in 10 seconds. Reference FSAR 8.2.3.1.b. The SAR goes on to state that the time delay for a LOOP and simultaneous LOCA is 36 seconds considering signal generation, electrical start up, and injection pump start up and initiation.

The basis document of 1107-3 contains a note on starting air pressure. It states: "The diesel generator has the ability to start and load with an air pressure as low as 100#. Based on physical condition of the air start system, the diesel generator can be considered in reduced availability and may not meet the 10 second start/load criteria. Contact system engineering to address operability under degraded conditions when below 175 psig."

The candidates recognized 120 psig as a pressure above the minimum pressure necessary for the diesel to meet the design function as opposed to the operability function and chose answer "A". The concern is that the question identifies specific air start pressure associated with the diesel air start system and provides a specific design basis condition to address. Given the LOOP and ESAS the question is asking if the diesel meets design basis condition. With a steady air pressure at 120 psig the diesel will meet the design condition. The question is not asking an operability determination but a determination of design basis.

Recommended resolution:

Change answer key to A.

Justification:

The question identifies specific criteria associated with the diesel air start system and a specific design basis event. The air start system is not indicated degraded in the question other than reduced and stable air pressure of 120 psig. With an intact air start system without further degraded conditions the diesel generator has been demonstrated and tested to start and be ready to load in less than 10 seconds at less than 100 psig air start pressure. The procedure change that initiated this note identified and used this successful test of both emergency diesel generators at less than 100 psig as a basis for change. The Plant Review Group at the time of change was concerned with an operability determination and not design basis function. PRG meeting minutes 1991-04 specified the diesel remained operable with a starting air pressure of

175 psig, based on startup testing conducted by TP 401/1(6 starts, with the last beginning at 175 psig.) Additional PRG meeting minutes agreed with the proposal to conduct a special test procedure to determine whether the diesel could become operable at even lower air pressures. PRG minutes 1991-013 reviewed the performed test and recognized the "A" diesel could start and load in less than or equal to 10 seconds with 75 psig air pressure. PRG meeting minutes from 1992-005 further reviewed and had engineering evaluate a starting air pressure of 100 psig. This evaluation resulted in the procedure change that was used in this question. It was evident the concern for a reduced air pressure was any further degradation from an air starting system standpoint between projected surveillance runs conducted monthly or potential degradation from the time of initial air system pressure identified to the time of real diesel demand. The 175 psig remained in the procedure as a given threshold for operability with the ability to evaluate down to 100 psig with a specific set of conditions. As a result of the manner in which this question is constructed, the specific conditions presented, the basis for the referenced note the correct choice is "A". The diesel will meet design function.

Attached References:

Procedure Change Safety Determination
FSAR 8.2.3.1.b
1107-3 Note
Engineering response to ILT exam issue

Question ID Number: #073

Concern or Problem:

The question references an Operations expectation that was recently changed and is after the freeze date established for the ILT class.

Recommended Resolution:

Accept either "B" or "D" based on original expectation, current expectation, and OS-24, "Conduct of Operations During Abnormal and Emergency Events".

Justification:

The procedure freeze date was clearly communicated to the students. However, this particular OPS expectation was communicated to the students only one week prior to the beginning of the NRC exam and some students disregarded the date on the question since they believed that they were to answer the question based on procedures in effect on the day of the procedure freeze.

Additionally, there was only a short time to train on the new OPS expectation before the exam date. The students were only in the simulator for two days following the date of the OPS expectation change, as compared to about 4 months of training with the original OPS expectation.

Attached References:

OS-24 , "Conduct of Operations During Abnormal and Emergency Events". Section 4.1.14,
Operations expectation 10/6/04
Operations expectation 05/02/05

Form ES-401-5 **Written Exam Question Worksheet** Q # 073

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.12 Page # 2-12 RO/SRO Importance Rating 3.4 3.9

Question Knowledge of general operating crew responsibilities during emergency operations.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ✓ 55.41 .10 55.43

Proposed Question ✓ **RO** **SRO** **PRA Related** **Correct Answer** B.

Identify the ONE selection below that describes procedure place keeping requirements during implementation of EMERGENCY OPERATING PROCEDURES (EOPs), IAW OS-24 "Conduct of Operations During Abnormal and Emergency Events", and operations expectations in effect on 5/16/05.

During EOP implementation ____________________ are REQUIRED to be checked or otherwise marked.

A. only transitions between procedures

B. only steps provided with check-off spaces

C. all EOP steps, whether or not check-off spaces are provided

D. all steps with check-off spaces, and ALL EOP Rule/Guide steps

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, section 4.1.14, Page 14, Rev. 10.
TMI Operations Expectation Database, Placekeeping SOS Response dated 10/06/2004.

Proposed Exam Reference None.

Learning Objective LP 11.2.01.513, Obj. 2.

Question Source ✓ **New** **Bank** **Question #**

Modified Bank **Parent Question #**

Question NRC Exam History

Question Cognitive Level ✓ **Memory/Fundamental Knowledge** **Comprehension/Analysis**

Discriminant Validity Statement

A INCORRECT because transition between procedures is not the only time procedures are required to be marked for place keeping.

Distracter is plausible because transition between procedures requires an announcement by the CRS and could be interpreted as the only time for procedure placekeeping.

B CORRECT because OS-24 Guidance is referenced in the most recent revision of Operations Expectations.

C INCORRECT because it does not address EOP-010 rules guides and graphs.

Distracter is plausible because place keeping in EOPs is an extremely important operator fundamental.

D INCORRECT. Operations expectations changed to say place keep IAW OS-24.

Comments Modified 5/11/05, Operation expectation changed, "B" now the correct answer.

4.1.14 Place keeping in an EVENT PROCEDURE

A. Check-off spaces are checked or otherwise marked after the action required by the step is completed. If the procedure is re-performed, additional marks are used.

B. Check-off spaces for VERIFY steps when used in two column format, are completed as follows. If the condition is satisfied, mark the space for the VERIFY step and leave the right hand column spaces blank. If the condition is not satisfied, leave the VERIFY space blank, and mark the spaces in the right hand column after the action required by the step is complete.

C. 24 Hour clock time should be entered in the TIME spaces which occur periodically throughout the EOP. These reference times are used to perform time dependent actions or to reconstruct the event.

D. EOP Rules posted on the Control Boards contain check-off spaces that are not required to be checked or otherwise marked as the step is performed by Reactor Operators. The check-off spaces are marked afterward as a verification that the Rule was performed correctly.

E. CARRYOVER STEPS are left blank until the step applies, and marked NA after the procedure is completed if the step condition was not satisfied.

4.1.15 TWO COLUMN Format

A. The user of the procedure reads the "ACTION/EXPECTED RESPONSE" from the left hand column.

B. If the action is completed satisfactorily or if the response is as expected, then the user proceeds down to the next step in the left hand column (and skips the right hand "Response not obtained" column)

C. If the <u>action</u> cannot be completed or the response is not as expected, then the user proceeds to the right hand column. The user takes the action described in the right hand column and proceeds to the next step in the left hand column.

D. If a "**VERIFY**" step is used in the LH column and no RNO is specified, then do not proceed past this step if the condition is not satisfied.

TMI Operations Expectation

The objective of the expectations process is to assure uniform application of expectations and standards in the plant and the training environment. The expectations process document provides a method for instructors, students and operations license holders to solicit clarification from the SOS when clarification from the employees immediate management does not resolve the issue. In the case of training instructors it is a direct method to obtain documented feedback on Operations Expectations and Standards. It can be a method to determine how to implement certain aspects of procedures but it is not a method to change procedures. Deficient procedures are addressed through IR process and enhancements are through direct discussion with the procedure owner.

Date: 10/06/2004

Originator: Ken McCall/TMI Extention: 2061

Description: Placekeeping is not consistently executed when ROs utilize the hardcards for Rules & Guides. Some provide checkoff lines and others do not. Some operators execute the guide then placekeep when verifying their actions.

Fundamental: Procedure Adherence Procedures

Recomendation: Placekeeping is performed on all Rules & Guides as they are performed.

Date: 10/06/2004

Response by SOS: Placekeeping shall be performed on all Rules & Guides as they are performed if there are no signoff lines the user should placekeep as well.

Feedback Mechanism:

Reply to:

- [x] TMI_SRO
- [ ] TMI_CRO
- [ ] TMI_AO
- [x] TMI_Training Ops Group

TMI Operations Expectation

The objective of the expectations process is to assure uniform application of expectations and standards in the plant and the training environment. The expectations process document provides a method for instructors, students and operations license holders to solicit clarification from the SOS when clarification from the employees immediate management does not resolve the issue. In the case of training instructors it is a direct method to obtain documented feedback on Operations Expectations and Standards. It can be a method to determine how to implement certain aspects of procedures but it is not a method to change procedures. Deficient procedures are addressed through IR process and enhancements are through direct discussion with the procedure owner.

Date: 10/06/2004

Originator: Ken McCall/TMI Extention: 2061

Title: Placekeeping expectation for hardcard Rules & Guides *provide a brief title of issue*

Description: Placekeeping is not consistently executed when ROs utilize the hardcards for Rules & Guides. Some provide checkoff lines and others do not. Some operators execute the guide then placekeep when verifying their actions.

Fundamental: Procedure Adherence Procedure: -

Recomendation: Placekeeping is performed on all Rules & Guides as they are performed.

Date: 05/02/2005

Response by SOS: Placekeeping shall be performed on all Rules & Guides as directed by OS-24.

Feedback Mechanism:

Reply to:

- [x] **TMI_SRO**
- [x] **TMI_CRO**
- [ ] **TMI_AO**
- [x] **TMI_Training Ops Group**

Since I was the one who approached Joe D'Antonio regarding Q# 073, I'll offer this response:

The revised "ops expectation" was conveyed to the students the same way all "ops expectations" initially get conveyed to the students, and that was via e-mail. Ops expectations aren't specifically called out in the simulator guides for simulator training. They get re-enforced throughout the program, but usually only as they apply for each given scenario.

Every candidate was notified of the change via e-mail. Some candidates participated in the discussion with the Shift Operations Superintendent (Randy Campbell) regarding the changed expectation. Not every candidate was in the room when Randy made the change, but every candidate received the e-mail which is automatically sent whenever Randy makes one of those changes. There was no formal training on the new OPS Expectation, but there is no "formal" training on *any* OPS Expectation.

The candidates weren't notified prior to the NRC prep week. The change wasn't made until 5/2/05, which was after the NRC prep week. If I mis-conveyed that to Joe D'Antonio, please extend my apologies. On the afternoon of the first simulator scenario - 5/9/05 - I told him that it had been recently changed, and that students had been informed of the change and further that it wasn't a "procedure" so I didn't know how the students would interpret the impact of a "procedure freeze" on something that's not a procedure. I believed the students would know that the expectation wasn't a "procedure" and so that if we made it clear, they would answer based on OS-24 alone. Obviously, I was wrong. Only one student answered based on current expectations and three answered based on the expectation in effect as of the procedure freeze date. These candidates answered based on the bulk of their training.

I can't provide any additional references, because the only references are within the two OPS Expectations and OS-24. OPS Expectations aren't specifically called out in the simulator guides for simulator training. They get re-enforced throughout the program, but usually only as they apply for each given scenario.

I don't know if this will answer all of Joe's questions, please let me know if he has any others.

Matt Smith

Question ID Number: #097

Concern or Problem:

Based on student feedback, "the wording of choice "D" is unclear. Some students believed the words "in support of" were **NOT** the equivalent of "as directed by." This interpretation would make choice "D" an additional correct answer."

Recommended resolution.

Accept "C" or "D" as correct.

Justification:

The assumption that "in support of" meant an activity performed outside of a surveillance activity, but supporting the activity by establishing necessary conditions is a reasonable assumption. One student questioned this wording specifically during the exam. No clarification was provided beyond "Do the best you can with the information given."

If specific instructions for implementation, removal and configuration restoration are not included in the surveillance, the installation of a jumper "in support of the surveillance" would clearly not be a pre-engineered activity. As such, a TCCP (Temporary Configuration Change Package) processed per CC-AA-112 would be required.

Attached References:

CC-AA-112 Attachment 2

Examination Outline Cross-Reference Tier # 3

Evolution/System Equipment Control Group #

K/A # 2.2.5 Page # 2-5 RO/SRO Importance Rating 1.6 2.7

Measurement Knowledge of the process for making changes in the facility as described in the safety analysis report.

10CFR55.43(b)(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .3

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** C.

Identify the ONE temporary change below that requires processing and approval using CC-AA-112, Temporary Configuration Changes.

A. Installation of rigging to support maintenance.

B. Installation of temporary lead shielding to reduce radiation dose.

C. Installation of an inflatable plug to seal a concrete pipe penetration.

D. Jumper installation to support performance of a surveillance procedure.

Technical Reference CC-AA-112, Temporary Configuration Changes, Attachment 2, Pages 24 and 25, Rev. 8.
Temporary Change Tracking Log item 04-00845.

Open Exam Reference None.

Learning Objective None.

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because installation of rigging to support maintenance is typically addressed by pre-engineered procedures (CC-AA-112 Page 24).

Distracter is plausible because it represents a temporary change to the plant.

B INCORRECT answer because installation plant barriers, including temporary lead shielding for reduction of radiation dose rates, is typically addressed by pre-engineered procedures (CC-AA-112 Page 24).

Distracter is plausible because it represents a temporary change to the plant.

C CORRECT answer.

D INCORRECT answer because jumper installation to support performance of a surveillance procedure is a repetitive action, typically controlled by the surveillance procedure itself (CC-AA-112 Page 25).

Distracter is plausible because it represents a temporary change to the plant.

Comments None.

ATTACHMENT 2
TCCPs, Exclusions and Associated Administrative Controls (CM 6.1.2.1& CM-6.1.5.3)
Page 1 of 3

Temporary configuration changes are controlled either through TCCPs or through use of procedures that have been pre-engineered. Pre-engineered procedures allow the Installer to place the detailed instructions for implementation, removal and configuration restoration directly into the work package used for performing the work without the need for a TCCP. Pre-engineered procedures are used to control changes that are performed on a regular basis (i.e. repetitive maintenance or repetitive repair) and would benefit from a more specifically detailed process. The criteria for use in developing new pre-engineered procedures is in Attachment 1 of CC-AA-112. If an approved pre-engineered procedure is not available for controlling a specific temporary change, then a TCCP is required. Activities controlled by pre-engineered procedures are therefore considered as "Exclusions".

Each station in Exelon may have pre-engineered procedures in place that are not available at other stations. Additionally, this procedure (CC-AA-112) identifies other Exclusions that have been agreed upon by all stations as activities that can be implemented without TCCPs. These Exclusions are listed in this Attachment. Various temporary changes are identified as Exclusions based on the simplicity of the change, and commonly acknowledged industry practices associated with performing day to day activities within the plant that do not have an impact on plant design based configuration.

Based on the above, the following table is provided to identify activities that typically require a TCCP, and a list of activities that are typically addressed by pre-engineered procedures. The actual determination of whether or not a specific activity can be performed as a TCCP or a pre-engineered activity depends upon what has been specifically approved for use at individual stations.

Controlled and Issued as TCCPs	Pre-Engineered Activities (See Note 1)
Temporary Setpoint Changes	Ventilation Dampers out of Normal Position (through Operations abnormal lineup procedure)
Mechanical jumpers (hose, tube, pipe) used as pressurized process flowpaths **(CM 6.1.6.3)**	Temp Lead Shielding
Valve Blocks Not Installed Within an Operations Clearance Boundary	Plant Barriers – includes Fire, Ventilation, Security, Radiation, Flood, High Energy Line Break, and Missile Barriers
Temp Power Feeds (TCCP unless Exclusion Item 6 applies)	Scaffolding mounted or attached to structures
Floor Drains with plugs installed	Procedure CC-AA-404 "Maintenance Specification: Application Selection, Evaluation and Control of Temporary Leak Repairs".
Pipe Supports	**Freeze Seals (CM-6.1.3.3)**
Lifted Leads / Pulled Circuit Boards **(CM 6.1.6.3)**	Rigging
Installed or Removed Filters or Strainer	
Gagged or Disabled Relief Valves **(CM 6.1.6.1)**	
Electrical Jumpers (is Maintenance developing a Maint. Alter. Procedure?) **(CM 6.1.6.3)**	
Disabled Alarm	
Battery Cell Jumpers **(CM 6.1.6.3)**	

ATTACHMENT 3
Temporary Configuration Change Precautions and Limitations
Page 1 of 2

1. Whenever possible, electrical circuits will be de-energized prior to the installation of jumpers or lifting of leads. If the TCCPs must be made with electrical circuits energized, specific approval of the Operations Supervisor is required. Consideration should be given to using fused or switched jumpers. The effects of arcing and electrical noise should also be considered during energized installations. **(CM-6.1.3.6)**

2. Lifted leads will be suitably insulated from other circuits and from ground.

3. Jumpers (not alligator clips or similar devices) installed during installation of the TCCP should be routed (tied off or taped) and/or should be of correct length (no loops or extra hanging wire) to prevent accidental dislodging or removal. Jumpers should also use ring lugs to prevent accidental dislodging or removal. Jumpers and power feeds that have ends which cannot be seen at the same time will have tags/cards at each end. **(CM-6.1.2.8, 6.1.3.4, and 6.1.3.5)**

4. If the proposed activity places portable equipment or hardware into the plant where it can impact/interact with plant SSCs, or circuits and is not controlled by other processes, then contact Engineering to evaluate the impact. Examples that may impact/interact with the plant are items that could cause: **(CM-6.1.3.6 & CM-6.1.5.7)**
   - Falling/Interaction
   - Initiation of a fire
   - Overheat
   - Explosion
   - Impairment of a FP zone
   - Additional loading on electrical circuits
   - Change in airflow or HVAC conditions
   - Change in, or impairment of fluid flows
   - Alteration, impairment, or creation of penetrations
   - Increase in dose, etc.
   - Introduction of foreign material in the drywell or containment that could become LOCA generated debris that may plug ECCS strainers or ECCS sump screens.

5. Do not cross-connect systems that are not specifically designed for cross-connection. When connecting the service air system to other systems which could lead to cross-contamination of the service air system <u>or</u> when connecting the demineralized water system to other systems which could cause contamination of the demineralized water system, appropriate controls shall be used (e.g., check valves) to ensure no backflow of contamination will occur.

6. The use of manually operated valves or manually operated pneumatic pressure regulators to control pressure in lieu of an automatic pressure regulator valve should be a short term alternative.

7. Jumpers or lifted leads should be utilized in lieu of non-conductive blocks to prevent relay contacts from changing state. **(CM-6.1.3.7)**

Question ID Number: #097

Concern or Problem:

Based on student feedback, "the wording of choice "D" is unclear. Some students believed the words "in support of" were **<u>NOT</u>** the equivalent of "as directed by." This interpretation would make choice "D" an additional correct answer."

Recommended resolution.

Accept "C" or "D" as correct.

Justification:

The assumption that "in support of" meant an activity performed outside of a surveillance activity, but supporting the activity by establishing necessary conditions is a reasonable assumption. One student questioned this wording specifically during the exam. No clarification was provided beyond "Do the best you can with the information given."

If specific instructions for implementation, removal and configuration restoration are not included in the surveillance, the installation of a jumper "in support of the surveillance" would clearly not be a pre-engineered activity. As such, a TCCP (Temporary Configuration Change Package) processed per CC-AA-112 would be required.

One candidate recognized both "C" and "D" as being correct and chose "D" because of the belief "D" had more significant impact on the CRS than "C". Although not typical, examples exist where TCCP's are used in support of surveillance procedures. One process area where this occurs is in the conduct of troubleshooting. The impacts of troubleshooting results in use of TCCP's in support of surveillances to meet the troubleshooting needs. This practice is evident in RR-V-6 diagnostic testing. ECR TM 03-00620 000 was used with 1300-3K to conduct diagnostic testing.

Attached References:

CC-AA-112 Attachment 2