FACILITY POST-EXAMINATION COMMENTS

FOR THE PRAIRIE ISLAND INITIAL EXAMINATION - AUGUST 2005

NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

AUG 2 5 2005

L-PI-05-078
10 CFR 55

Mr. Dell McNeil
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2443 Warrenville Road, Suite 210
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Facility Recommendations for Prairie Island Nuclear Generating Plant (PINGP) Reactor Operator (RO) and Senior Reactor Operator (SRO) License Examination Changes

Nuclear Management Company, LLC (NMC) has completed its grading and review of the RO and SRO license written examinations conducted the weeks of August 8 and August 15, 2005 at PINGP. The grading and review was performed in accordance with NUREG-1021, ES-403, "Grading Initial Site-Specific Written Examinations" and as a result we recommend deleting one question and changing three questions on the RO exam.

Attached to this letter are the facility recommendations describing the recommended examination changes with associated references.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant

Enclosure (1)

cc: Regional Administrator, USNRC, Region III, w/o enclosure
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AUG 2 6 2005

ENCLOSURE 1

Prairie Island Nuclear Generating Plant
Facility Recommendations for RO and SRO License Examination Changes

RO 4	(6 pages)
RO 11	(8 pages)
RO 25	(4 pages)
RO 38	(9 pages)
RO 54	(4 pages)
RO 67	(5 pages)
RO 71	(6 pages)

42 pages follow

Question : RO 4

Comment:

USAR Section 14.6 Table 14.6-3 assumption 2C states "assuming offsite power available." The note states "studies show that continued operation of the reactor coolant pumps results in worse peak cladding temperature." Thus, offsite power must be maintained as described in the question. Table 14.6-4 timeline and the description of page 14.6-3 show SI injection to the cold legs at 12 sec. and upper plenum injection (RHR) at 22 sec. Since the blowdown phase did not end until 21.5 sec., and the bottom of core uncovery starts at 33.5 sec., RHR is injecting during the refill stage.

Facility Response:

There are conflicting references regarding this question. B18B, Emergency Core Cooling System, Section 4.3.C states that refill is accomplished by the accumulators. The above referenced USAR section gives a timeline showing that RHR is injecting during the refill stage. A Procedure Change Request has been submitted to change B18B to refer to the current LBLOCA analysis.

Based on these conflicting references, the site **recommends accepting both answers "A" and "C".**

4. During a Large Break LOCA, all ECCS flow is assumed to bypass the core until the completion of the Blowdown Phase. During the Refill Phase immediately following blowdown, the ECCS flow is directed to the ________.

   a. cold legs AND reactor vessel simultaneously to refill the core from the top and bottom at the same time.

   b. reactor vessel ONLY as complete core uncovery occurs during blowdown and core injection is the most effective cooling method.

   c. cold legs ONLY to refill the core barrel and start the recovery of the core from the bottom up.

   d. cold legs AND hot legs simultaneously to ensure either SI or Accumulator injection will pass through the core on the way to the break.

ANSWER: C

Explanation:	**A: Incorrect, RHR flow requires a minimum of 15 seconds after SI to start (based on safeguards load sequencers) and it is the only ECCS source normally aligned to the reactor vessel.** **B: Incorrect, the initial response is via Accumulators (passive) and SI (no start delay on safeguard load sequencers) and both of these inject via the cold legs.** **C: Correct, per above.** **D: Incorrect, no ECCS source discharges to the hot leg. Plausible as train separation ensures at least one train passes through core on the way to the break.**
Technical References:	**Fig B18A-1, B18A-5**
Objective:	**P8180L-005, Objective 4f**
KA Statement:	**Knowledge of the reasons for the following as they apply to the Large Break LOCA: Injection into cold leg**

Cognitive Level:	**1-F**	**10CFR55.41**	**8**	**10CFR55.43**		**New Question**	**Yes**
Bank ID:		**Question ID:**		**Modified:**		**Last NRC Exam:**	

4.3. LOSS OF COOLANT ACCIDENTS

A. RCS LEAKS

Loss-of-coolant accidents are categorized based on the size of the break. Breaks of less than 3/4" equivalent diameter are considered to be RCS leaks. The charging pumps and normal makeup can maintain the RCS inventory for breaks of this size. If the leak is less than the Technical Specification limit or if the leak can be isolated, the unit can remain at power.

B. SMALL BREAK LOSS OF COOLANT ACCIDENTS

Breaks of $\geq$ 3/4" to 12-1/2" (1 ft^2) equivalent diameter are considered to be small break LOCAs (SBLOCAs). The loss of coolant caused by this size of break cannot be handled by the normal charging system. The RCS depressurizes and a 'S' signal is generated, activating the ECCS. The SI pumps and, if pressure falls below ~730 psig, the accumulators provide the makeup required by the RCS to maintain the inventory.

C. LARGE BREAK LOSS OF COOLANT ACCIDENTS

Breaks of 1 ft^2 to a double ended rupture of a RCS pipe are considered to be large break LOCAs. The large break LOCA is the design basis for much of the ECCS.

A large break LOCA has the following four characteristic stages:

- Blowdown - Blowdown begins with the initiation of the LOCA and ends when the RCS pressure falls to that of the containment atmosphere. During blowdown, the RCS undergoes a significant loss of water mass. All ECCS flow bypasses the core during blowdown stage.
- Refill - Refill begins at the end of blowdown and ends when the addition of ECCS water fills the bottom of the reactor vessel up to the bottom of the fuel rods. Refill is accomplished by accumulator injection.
- Reflood - Reflood is the time from the end of refill until the reactor vessel has been filled with water such that the core temperature rise has been stopped. Reflood is initiated by continued accumulator injection and accomplished by SI and RHR pump injection. The RHR portion of the ECCS is primarily responsible for core reflood. After successful reflood, ECCS injection continues until the RWST is depleted.

located under an open hole in the upper core plate, which was shown in sensitivity studies to be the limiting location for peak cladding temperature. These transients were considered to be terminated if the hot rod cladding temperature began to decline and the injected ECCS flows exceeded the break flow.

14.6.4 Description of a Nominal Large Break LOCA Transient

Table 14.6-3 shows the time sequence of events and a summary of the important results. Figures 14.6-1 through 14.6-32 present various plotted quantities from the Addendum 4 Case with Higher Core Limits WCOBRA/TRAC calculation. A discussion of the computed transient follows. Table 14.6-4 contains the time sequence of events and other useful information, such as peak cladding temperatures and burst information, related to understanding the figures shown for the computed Addendum 4 Case with Higher Core Limits transient results. The figures for PCT do not reflect subsequent error discovery or change notifications.

The following discusses some of the key transient phases and reference to the above plots is used to illustrate the description. However, in instances where several plots may illustrate the same behavior, the most descriptive figure(s) for illustrating the point will be used.

As in the Addendum 4 Base Case, shortly after the break opens, the vessel rapidly depressurizes (Figure 14.6-4) and the core flow quickly reverses (Figures 14.6-9, 11, 13, 14, 17, 18, 20, and 21. During the flow reversal, the hot assembly fuel rods dry out and begin to heat up momentarily (Figures 14.6-16 and 14.6-1 through 14.6-3). At approximately 12 seconds into the transient, maximum downflow is reached in the high and low power regions of the core. As the vessel continues to depressurize, liquid inventory continues to be depleted (Figure 14.6-23), and core void fractions increase (Figures 14.6-12, 15, 19 and 22). This results in reduced core flow and resulting cladding heatup in all regions of the core (Figure 14.6-1 through 14.6-3).

At approximately 8 seconds into the transient, the accumulator begins to inject water into the intact cold leg (Figure 14.6-5). This water fills the cold leg and upper downcomer region, and is bypassed to the break initially. At approximately 12 seconds, pumped injection into the cold leg begins (Figure 14.6-7), and at approximately 22 seconds, pumped safety injection into the upper plenum begins (Figure 14.6-6). At approximately 30 seconds, accumulator water begins to flow into the lower plenum (Figures 14.6-23 through 14.6-25). The upper plenum injection begins to flow through the low power peripheral region of the core (Figure 14.6-21), and contributes to some core cooling, but primarily flows through the core into the lower plenum. The remaining rods experience a heat up as the accumulator and safety injection flows accumulate in the low plenum (Figures 14.6-1 through 14.6-3, and 14.6-23).

TABLE 14.6-3 ASSUMPTIONS USED IN THE APPENDIX K CALCULATIONS

1. PLANT CONFIGURATION
   a. Pressurizer in Intact Loop
   b. Total Peaking Factor (F_Q) of 2.40
   c. Nuclear Enthalpy Rise Peaking Factor ($F_{\Delta H}$) of 1.77 (Higher Core Limits Case)
   d. Hot Assembly Average Rod Power Factor (P_{HA}) of 1.59 (Higher Core Limits Case)
   e. Core Power of 102% of 1650 MWt
   f. Steam Generator Tube Plugging level of 15%
   g. Best Estimate Loop Flow Rate of 93,700 gpm
   h. Beginning of Cycle Fuel Temperature
   i. Beginning of Cycle Fuel Pressure
   j. Conservative power distribution

2. SAFETY INJECTION CONFIGURATION
   a. Worst single failure
   b. One High Head Safety Injection line Spilling to Containment Pressure
   c. Maximum Safety Injection Delay Time (assuming Off-Site Power available) [1]

3. MODEL ASSUMPTIONS
   a. Accumulator Nitrogen Modeled
   b. Conservative Reactor Coolant Pump Two-phase Multiplier
   c. Cross-flow De-entrainment In the Upper Plenum
   d. Limiting Break Discharge Coefficient (C_D) of 0.4
   e. Peak Linear Heat Rate of 15.167 kw/ft
   f. Lower Bound Containment Pressure
   g. Decay Heat with ANSI/ANS 5.1 1971 Standard +20%
   h. Baker-Just Correlation for Metal-Water Reaction Calculation

(1) Sensitivity studies show that continued operation of the Reactor Coolant Pumps results in worst peak cladding temperature.

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TABLE 14.6-4 ADDENDUM 4 CASE WITH HIGHER CORE LIMITS RESULTS SUMMARY

EVENT	TIME	
Start of Accident	0.00	
Reactor Trip Signal	2.23	
Safety Injection Signal	2.23	
Accumulator Injection Begins	8.00	
Safety Injection Begins	12.23	
End-of-Bypass	19.72	
End-of-Blowdown	21.51	
Bottom-of-Core Recovery	33.48	
Intact Loop Accumulator Empty	38.76	
Peak Cladding Temperature Occurs	60.00	
RESULTS	**VALUE**	
Blowdown PCT (°F)	1846	
Blowdown PCT Time (sec)	9.00	
Blowdown PCT Location (ft)	6.625	
Bottom-of-Core PCT (°F)	1895	
Bottom-of-Core Recovery Time (sec)	33.94	
Bottom-of-Core PCT Location (ft)	7.250	
Accumulator Empty PCT (°F)	1969	
Accumulator Empty Time (sec)	38.76	
Accumulator Empty PCT Location (ft)	7.625	
Total Reflood PCT (°F)	2180 (2002, Ref. 33)	01-100
Total Reflood PCT Time (sec)	60.00	
Total Reflood PCT Location (ft)	7.625	
Hot Rod Burst Time (sec)	20.58	
Hot Rod Burst Location (ft)	7.753	
Hot Assembly Burst Time (sec)	24.82	
Hot Assembly Burst Location (ft)	7.753	
Hot Assembly Blockage (%)	32.23	
Maximum Local Zirc-Water Reaction (%)	5.882	
Elevation of Maximum Zirc-Water Reaction (ft)	7.625	
Total Zirc-Water Reaction (%)	< 1.0	

Question : RO 11

Comment:

Reference NUREG 1021, Appendix E Part B.7 states not to assume conditions that are not specified in the stem of the question. Due to the fact we are trained to assume a normal lineup makes "C" and "D" correct.

Facility Response:

A normal Pressurizer Heater Lineup would have Bank A powered from a safety-related source and banks B, D and E from a non-safeguard source that would be deenergized based on stem conditions.

ERCS display "M1.97" displays the power (KW) supplied to A and B heater banks. If a normal lineup is assumed, the operator could verify power to Bank A while knowing power is lost to banks B, D and E. This would make answer "C" correct.

Since the ERCS display will show the status of banks A and B regardless of electrical lineup, and states correctly that D and E Banks are deenergized regardless of breaker indication in the second part, "D" is also correct.

Recommend accepting both answers "C" and "D".

Level	RO	Tier	1	Group	1	K/A#	056 AA2.17	Imp. RO	3.0	Imp. SRO

11. All offsite power has been lost and safeguards buses are being supplied by their respective diesel generators.

    How can the status of each Pressurizer Backup Heater Bank be determined from the Control Room?

    a. Check for RED light indication on the associated Heater Bank Control Switch; if LIT, the bank is ENERGIZED.

    b. Check for RED light indication on the Bank A and B Heater Bank Control Switches; if LIT, the bank is ENERGIZED.
       Banks D and E are <u>NOT</u> energized regardless of control switch indication.

    c. Use the ERCS M1.97 display to view the power supplied to Bank A heaters.
       Banks B, D and E are <u>NOT</u> energized regardless of control switch indication.

    d. Use the ERCS M1.97 display to view the power supplied to Bank A or Bank B heaters.
       Banks D and E are <u>NOT</u> energized regardless of control switch indication.

ANSWER: D

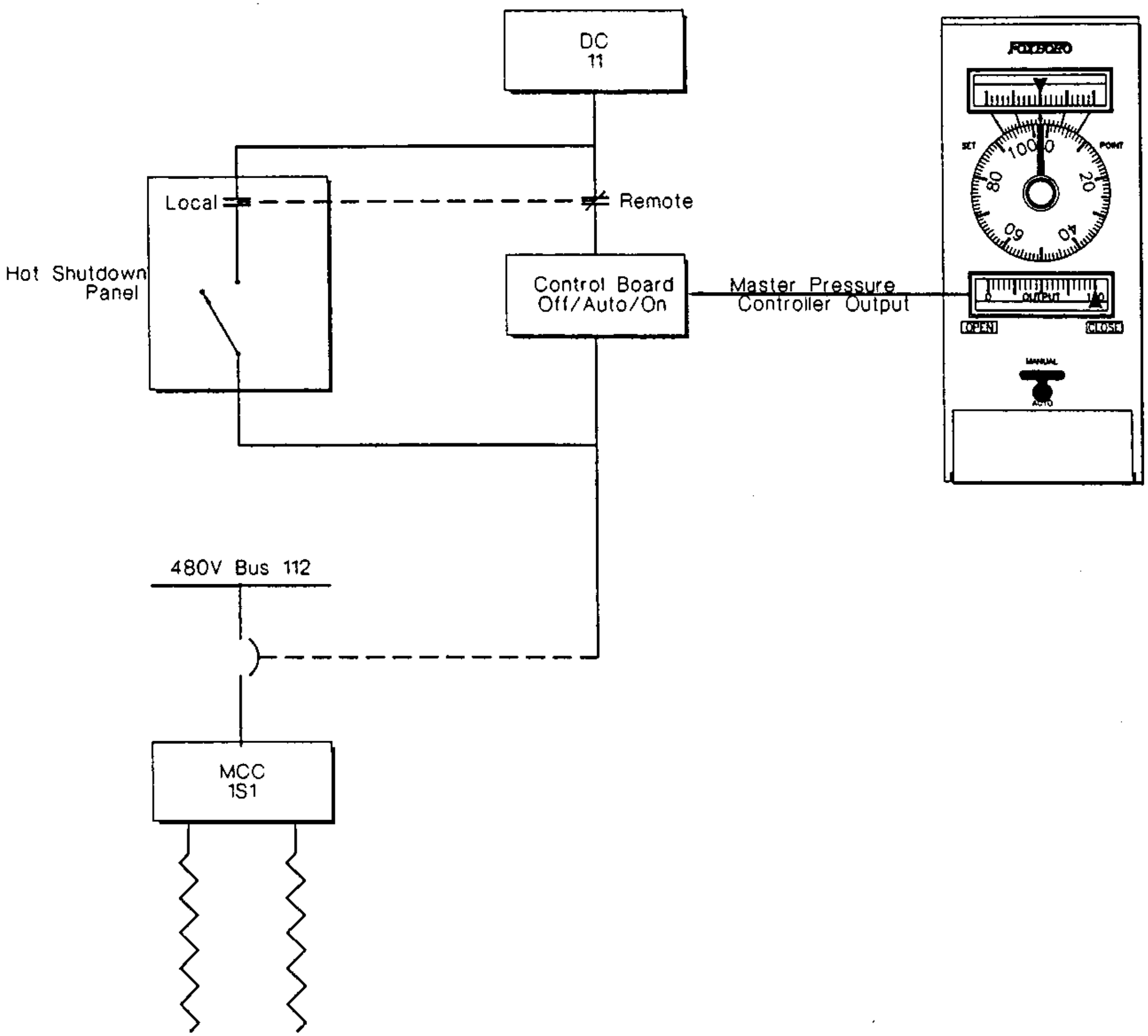
Explanation:
A: Incorrect, the control switches indicate breaker position only, not whether the associated MCC is powered. Banks D and E are supplied from non-safeguards power.
B: Incorrect, Bank B can be supplied from either a safeguards or non-safeguards (normal) supply, so breaker position is not evidence of being powered even with the stem conditions given.
C: Incorrect, Bank B could be energized if it is aligned to its alternate (safeguards) source.
D: Correct, the ERCS system calculates the KW loading of the A and B heater banks as they are/can be supplied from safeguards power.

Technical References:
ERCS M1.97 display
Fig B20.6-7b
Dwg PZP-026
MCC Report 1P2, 1R2

Objective: **P8170L-005 #3, 7a**

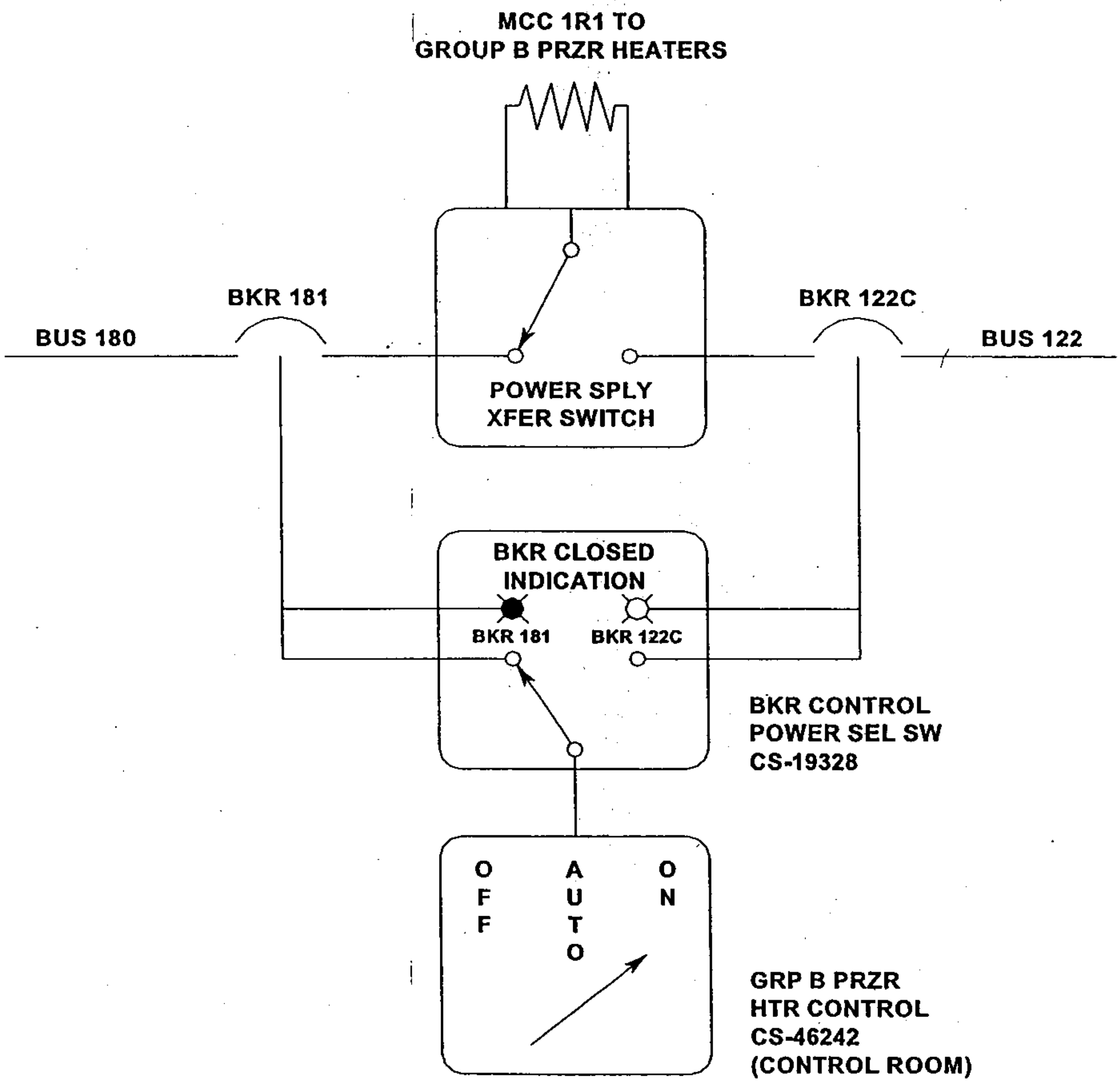
KA Statement: **Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of Pressurizer Backup Heaters**

Cognitive Level:	1-F	10CFR55.41	7	10CFR55.43		New Question	Yes
Bank ID:		Question ID:		Modified:		Last NRC Exam:	



Northern States Power Company **Prairie Island Training Center**	**Pressurizer Heater Group A**	**REV: 1** **10/11/94**	**PZP-026**

FIGURE B20.6-7b - GROUP B PRESSURIZER HEATER DIAGRAM



GROUP B PRESSURIZER HEATER DIAGRAM

B20.6 480V STATION AUXILIARY SYSTEM

1. SYSTEM FUNCTION

The function of the 480 VAC Station Auxiliary System is to provide a reliable source of power to low voltage auxiliary equipment. This equipment may be required during any one of the normal or emergency modes of plant operation.

2. SYSTEM GENERAL DESCRIPTION

2.1. 480 VAC Auxiliary Non-Safeguard System

Seven power centers make up the Non-Safeguard portion of the system. Five of these centers are arranged in a split bus configuration (double-ended). Each bus section has an associated load break disconnect switch and a 4160/480 volt transformer. The buses may be joined by use of a bus tie breaker. The double-ended centers are made up of the following bus pairs: 130/140, 150/160, 230/240, 250/260, and 190/290. Busses 180 and 270 single busses (single-ended) with associated "load break disconnect" switches and 4160/480 volt transformers. The non-safeguard buses receive power from 4160 VAC buses 13, 14, 23, and 24. Figure B20.6-1 illustrates the layout of the 480 VAC Non-Safeguard Distribution System.

2.2. 480 VAC Auxiliary Safeguards System

Eight 480V AC buses make up the Safeguards portion of the 480 VAC auxiliary system. The buses for Unit 1 are numbered 111, 112, 121 and 122. The buses for Unit 2 are numbered 211, 212, 221 and 222. Each bus has an associated 4160/480 volt transformer. Buses 111, 112, 211 and 212 provide power to the "A" Train safeguards loads and buses 121, 122, 221 and 222 provide power to the "B" Train safeguards loads.

The 4160V AC Safeguards buses 15 and 16 supply power to the Unit 1 480V AC safeguards buses. The 4160V AC Safeguards buses 25 and 26 supply power to the Unit 2 480V AC safeguards buses. Figure B20.6-2 illustrates the basic layout of the 480 VAC Safeguards Distribution System.

REVISION DATE: 09/04/03 ***** M C C REPORT *****
****

EQUIPMENT: MCC 1R2

EQUIPMENT NAME: MOTOR CONTROL CENTER 1R BUS 2
MAINT AREA: RODDR BUILDING: TURB FLOOR: 735 ROOM:
EQUIPMENT LOC : H.2/3.7 U1 ROD DRIVE RM
DRAWINGS

NF-40040
NF-40215

bus 180

SOURCE ID

EQUIPMENT ID : BKR 182
EQUIPMENT NAME: MCC 1R2 (PRZR HTRS GRP E)
MAINT AREA: RODDR BUILDING: TURB FLOOR: 735 ROOM:
EQUIPMENT LOC : G.4/5.8 U1 ROD DRIVE ROOM

LOAD ID

EQUIP ID	EQUIP ALT ID	EQUIP NAME
BKR 182-3	MCC 1R2-B10	PRZR HTRS 1-34;1-62&1-63
BKR 182-4	MCC 1R2-B11	PRZR HTRS 1-39;1-68&1-69
BKR 182-5	MCC 1R2-B12	PRZR HTRS 1-44;1-75&1-76
BKR 182-1	MCC 1R2-B8	PRZR HTRS 1-24;1-51&1-52
BKR 182-2	MCC 1R2-B9	PRZR HTRS 1-7;1-29&1-57

REVISION DATE: 09/04/03 ***** M C C REPORT *****
****

EQUIPMENT: MCC 1P2

EQUIPMENT NAME: MOTOR CONTROL CENTER 1P BUS 2
MAINT AREA: RODDR BUILDING: TURB FLOOR: 735 ROOM:
EQUIPMENT LOC : H.1/3.7 U1 ROD DRIVE RM
DRAWINGS

NF-40040
NF-40215

Bus 180.

SOURCE ID

EQUIPMENT ID : BKR 184
EQUIPMENT NAME: MCC 1P2 (PRZR HTRS GRP D)
MAINT AREA: RODDR BUILDING: TURB FLOOR: 735 ROOM:
EQUIPMENT LOC : G.4/5.8 U1 ROD DRIVE ROOM

LOAD ID

EQUIP ID	EQUIP ALT ID	EQUIP NAME
BKR 184-4	MCC 1P2-C10	PRZR HTRS 1-14;1-37&1-66
BKR 184-5	MCC 1P2-C11	PRZR HTRS 1-18;1-42&1-72
BKR 184-6	MCC 1P2-C12	PRZR HTRS 1-20;1-46&1-78
BKR 184-1	MCC 1P2-C7	PRZR HTRS 1-1;1-22&1-49
BKR 184-2	MCC 1P2-C8	PRZR HTRS 1-5;1-27&1-55
BKR 184-3	MCC 1P2-C9	PRZR HTRS 1-10;1-32&1-60

SELECT FUNC. KEY OR TURN-ON CODE [redacted] MAY 24, 2005 14:16:13 [redacted]

GROUP DISPLAY

M197 MISC. R.G. 1.97 PARAMETERS PAGE 1 OF 2
5 SECOND UPDATE RATE

POINT ID	DESCRIPTION	CURRENT VALUE	ENGR UNIT	ALARM LIMIT	QUAL CODE
1R0053A	SI PUMP AREA RAD LVL	0.35	MR/HR		GOOD
1R0054A	CS PUMP AREA RAD LVL	1.34	MR/HR		GOOD
1R0055A	AUX BLDG 695 EAST AREA RAD LVL	0.11	MR/HR		GOOD
1R0056A	AUX BLDG 695 WEST AREA RAD LVL	0.09	MR/HR		GOOD
1R0057A	AUX BLDG 715 EAST AREA RAD LVL	0.09	MR/HR		GOOD
1R0058A	AUX BLDG 715 WEST AREA RAD LVL	0.12	MR/HR		GOOD
1R0059A	AUX BLDG 715 PENET/LTDN AREA RAD	0.40	MR/HR		GOOD
1R0060A	AUX BLDG 735 NORTH AREA RAD LVL	0.08	MR/HR		GOOD
1R0061A	A STM LINE AREA RAD LVL	0.24	MR/HR		GOOD
1R0062A	AUX BLDG 755 EAST AREA RAD LVL	0.09	MR/HR		GOOD
1R0063A	AUX BLDG 755 WEST AREA RAD LVL	0.12	MR/HR		GOOD
1R0064A	TURB BLDG 735 NORTH AREA RAD LVL	0.11	MR/HR		GOOD
1R0065A	OPER SUPPORT CENTER RAD LVL	0.09	MR/HR		GOOD
1R0066A	D1 DSL GEN ROOM RAD LVL	0.10	MR/HR		GOOD
1N0053A	N51 NEUTRON FLUX SOURCE RNG LVL	0.0000+00	CPS		BAD
1N0054A	N52 NEUTRON FLUX SOURCE RNG LVL	0.0000+00	CPS		BAD
1N0055A	N51 NEUTRON FLUX LOG PWR RNG LVL	9.7657+01	LOG %		GOOD
1N0056A	N52 NEUTRON FLUX LOG PWR RNG LVL	1.1821-02	LOG %		GOOD
1N0057A	N51 NEUTRON FLUX LINEAR PWR RNG	99.8	%		GOOD
1N0058A	N52 NEUTRON FLUX LINEAR PWR RNG	99.7	%		GOOD
1Q0480A	SFGD PRZR BACK-UP HTRS GRP A PWR	0.0	KW		GOOD
1Q0481A	SFGD PRZR BACK-UP HTRS GRP B PWR	0.0	KW		GOOD
1T1000A	CNTMT AIR TEMP ELEV-697 35	83.7	DEGF		GOOD
1T1001A	CNTMT AIR TEMP ELEV-738 35	93.9	DEGF		GOOD
1T1002A	CNTMT AIR TEMP ELEV-755 35	99.8	DEGF		GOOD
1F0922A	SI FLOW TO COLD LEGS	0.0	GPM		GOO*
1F0923A	SI FLOW TO RX VSL	0.0	GPM		GOO*
1L2401A	COND STG TNK LEVEL 723	22.6	FT		GOOD
1L2402A	COND STG TNK LEVEL 724	22.5	FT		GOOD
1L0920A	RWST LEVEL 920	97.6	%	97.5	HALM

F1= F2= F3= F4= F5= LIBRARY F6= NEW GROUP
KBD= NORMAL AMODE= 1-FULL POWER U1-A*

Question : RO 25

Comment:

RO candidates are not required to memorize sequence of steps after transitions and if SI is reset or not. If it was stated SI was reset in the stem then I would know "C" was the only acceptable answer. Recommend accepting "A" and "C" as correct.

Facility Response:

RO candidates should know the major actions involved in an EOP, including the steps taken to establish an operator-controlled cooldown to the condenser. SI is reset prior to establishing the cooldown as it is preferred to use the Condensate and Feedwater system for steam generator level control, and these pumps cannot be operated until SI is reset. SI will be reset even if these pumps are not used or available.

Recommend no change to this question.

Level RO Tier 1 Group 2 K/A# E07 EA1.1 Imp. RO 3.6 Imp. SRO

25. Given the following:

- 1ECA-3.2 SGTR WITH LOSS OF REACTOR COOLANT: SATURATED RECOVERY is in progress.
- RCS T_{ave} is 552F and lowering.
- 12 SG is isolated with level 65% NR, rising at 2%/minute.
- Cooldown of the RCS is in progress using the Condenser Steam Dumps from 11 SG.
- A 95°F/hr cooldown rate has been established using steam dump MANUAL control.
- 11 SG steam flow is 0.53×10^6 lbm/hr.
- Pressurizer level is 30% and rising at 3%/minute.
- NO further operator action is taken.

Which ONE of the following conditions will occur FIRST assuming current trends continue?

a. 11 MSIV automatically closes.

b. 12 SG level goes offscale high.

c. Steam Dump flow is lost.

d. Pressurizer fills water solid.

ANSWER: C

Explanation:	**A: Incorrect, SI is already reset prior to entry into ECA-3.2 so will not occur. Plausible as SI + Lo Lo Tavg (540°F) + Hi Stm Flow (.505E6) will give 11 MSIV closure signal.** **B: Incorrect, (100-65%)/(2%/min)=17.5 minutes until this occurs.** **C: Correct, at 540°F the steam dump arming signal will be lost until BYPASS INTERLOCK position is momentarily selected on each train, and this will occur in (552-540°F)/(95°F/hr)(hr/60min)=7.6 min** **D: Incorrect, (100-30%)/(3%/min)= 23 min to upper range, plus time to go solid**
Technical References:	**ARP 47011-0203** **2ECA-3.1, 2ECA-3.2**
Objective:	***P8140S-200* Att 43, *Objective 8a***
KA Statement:	**Ability to operate and/or monitor the following as they apply to the (Saturated Core Cooling): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.**

Cognitive Level: 3-PEO 10CFR55.41 7 10CFR55.43 New Question No
Bank ID: DC Cook Question ID: E07EA1.1 Modified: Yes Last NRC Exam: 9/01

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Caution

- *IF RWST level decreases to less than 33%, THEN 1ES-1.2, TRANSFER TO RECIRCULATION, should be performed.*
- *IF offsite power is lost after SI reset, THEN manual action may be required to restart safeguard equipment.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Reset SI**	
2	**Reset Containment Isolation**	
3	**Establish Instrument Air To Containment**	
4	**Verify All AC Buses - ENERGIZED BY OFFSITE POWER**	Attempt to restore offsite power. IF offsite power NOT restored, THEN perform the following: a. Verify safeguard loads loaded on safeguard buses. b. Verify 92 kw diesel capacity available to run each charging pump.
5	**Check If Containment Spray Should Be Stopped:**	
	a. Spray pumps - RUNNING	a. Go to Step 6.
	b. Containment pressure - LESS THAN 20 PSIG	b. WHEN containment pressure less than 20 psig, THEN do Steps 5c and 5d. Continue with Step 6.
	c. Reset containment spray signal	
	d. Stop containment spray pumps	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Initiate RCS Cooldown To Mode 5, Cold Shutdown:**	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR	
	b. Use RHR System, if in service	
	c. Check one condensate pump - RUNNING	c. Start one condensate pump.
	d. Dump steam to condenser from intact SG	d. Manually or locally dump steam from intact SG using SG PORV. <u>IF</u> no intact SG available, <u>THEN</u> perform the following: • Use faulted SG. -OR- • If RHR <u>NOT</u> in service, <u>THEN</u> dump steam from ruptured SG.

Question : RO 38

Comment:

The bases for TS 3.6.5 Containment Cooling Systems under the LCO section states one train of CS and one train of CFCUs are required as a minimum for a SLB (see attached). Also, above 23 psig containment pressure, the crew would be in FR-Z.1 which ensures all CFCU's are operating and CL pressure is adequate. Of the selections given, "C is correct". Accept both "C" and "D" due to conflicting references.

Facility Response:

TS 3.6.5 states the requirement for one train of Containment Spray (CS) and one train of Containment Fan Coil Units (CFCU) for a Steam Line Break. The crew would be in FR-Z.1 given the current conditions. Answer D states one train of CS is adequate, and other than "B" CS, no other failures are given. With this information, answer D is correct. Additionally, answer C states that all four CFCU's are operating in Slow with full Cooling Water flow. Since it can be inferred from the stem that A train CS is working properly, C is also correct.

Recommend accepting answers "C" and "D" as correct.

Level	RO	Tier	2	Group	1	K/A#	022 A1.02	Imp. RO	3.6	Imp. SRO

38. Given the following on Unit 1:

- The Unit was at 100% power.
- A steam line break occurred in Containment.
- The reactor and turbine tripped.

The following conditions are noted:

- Containment pressure is 28 psig and increasing.
- B Train Containment Spray failed to actuate automatically or manually.

What action (if any) is required to prevent exceeding Containment design pressure limits?

a. Locally start 12 Containment Spray Pump and manually open the discharge valve.

b. Reset Containment Spray and stop Train A Containment Spray.

c. Verify all four CFCU's are operating in Slow with full Cooling Water flow.

d. None, one train of Containment Spray is adequate.

ANSWER: D

Explanation:	**A: Incorrect, one train of CS is 100% capacity and local operation is not directed.** **B: Incorrect, Containment pressure exceeds 23 psig therefore it is required.** **C: Incorrect, both Trains of CFCU's are not required. Plausible as TS used to say 1 CS pump equals 2 CFCU's.** **D: Correct, One train of CS is adequate to prevent exceeding design pressure.**
Technical References:	**Basis TS 3.6.5**
Objective:	**P8180L-002, Objective 7b**
KA Statement:	**Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:** **Containment pressure**

Cognitive Level:	3-PEO	10CFR55.41	8	10CFR55.43	New Question	Yes
Bank ID:		Question ID:		Modified:	Last NRC Exam:	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Containment Isolation Valves - CLOSED	IF flow path NOT necessary, THEN close valves.

Caution *IF 1ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION has been implemented, THEN containment spray should be operated as directed in 1ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION rather than Step 2 below.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Verify Containment Spray - RUNNING	
	a. Check containment spray - ACTUATED	a. Acutate containment spray.
	b. Check containment spray pumps - RUNNING	b. Start spray pumps.
	c. Check containment spray system valves - PROPER EMERGENCY ALIGNMENT	c. Align valves, as appropriate.
3	Check Containment FCUs - RUNNING in SLOW to the DOME	Align FCUs SLOW to the DOME.
4	Check Cooling Water Pressure, Loop A AND Loop B - GREATER THAN 65 PSIG	Restore cooling water pressure per C35 AOP1, LOSS OF PUMPING CAPACITY OR SUPPLY HEADER WITH SI.
5	Verify MSIVs And Bypass Valves - CLOSED	Close valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Caution

- *At least one SG must be maintained available for RCS cooldown.*
- *IF both SGs are faulted, THEN at least 40 gpm feed flow should be maintained to each SG.*

6	**Check If Feed Flow Should Be Isolated To Any SG:**	
	a. Check SG pressures: • ANY SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER -OR- • ANY SG COMPLETELY DEPRESSURIZED	a. Go to Step 7.
	b. Isolate feed flow to affected SG: • Main FW • AFW	
7	**Return To Procedure And Step In Effect**	

-END-

BASES

APPLICABLE SAFETY ANALYSES (continued)	Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time incorporates delays to account for load restoration and motor windup (Ref. 3). The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	During a LOCA or SLB, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and thereby maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs. Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon a containment spray actuation signal. Manual valves in this system that could, if improperly positioned, reduce the spray flow below that assumed for accident analysis, are blocked and tagged in the proper position and maintained under administrative control. Containment Spray System motor operated valves, MV-32096 and MV-32097 (Unit 1), and MV-32108 and MV-32109 (Unit 2) are closed with the motor control center supply breakers in the off position. Each Containment Cooling System typically includes cooling coils, dampers, fans, and controls to ensure an OPERABLE flow path.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a LOCA or SLB could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

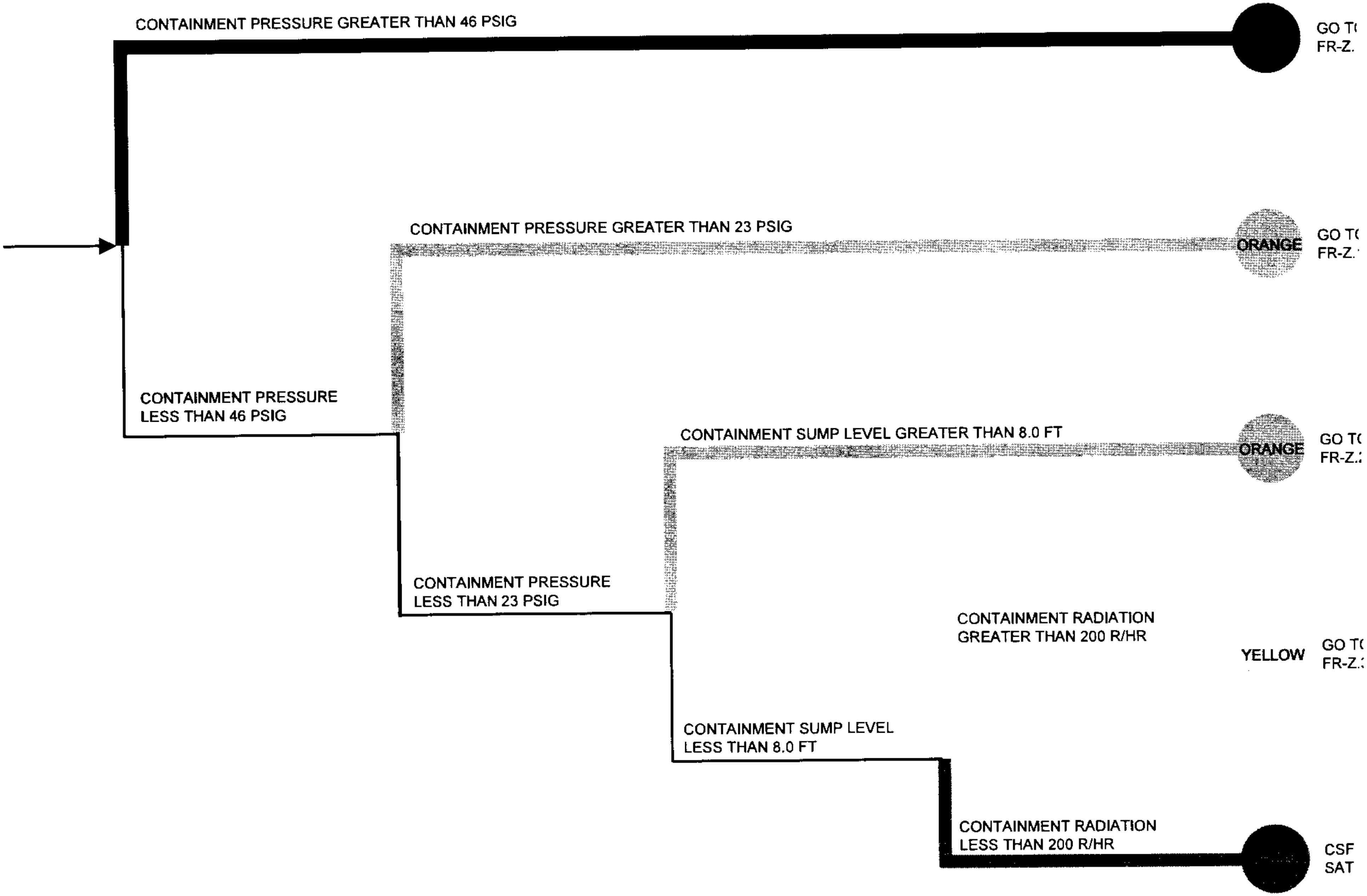
ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the other Containment Spray train, reasonable time for repairs, and low probability of a LOCA or SLB occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

F-0.5	CONTAINMENT	REV. 4



BACKGROUND INFORMATION FOR

1F-0.5, CONTAINMENT STATUS TREE

This status tree provides a systematic method to explicitly determine the status of the Containment Critical Safety Function. The intent of the Containment Safety Function is to maintain containment integrity, since this represents the third and final barrier against radioactivity release. In order to evaluate the status of this Critical Safety Function, the tree evaluates several possible challenges to containment integrity or essential equipment inside containment and directs the operator to an appropriate procedure for function restoration. The function is satisfied if containment pressure is below 23 PSIG, containment level is less than flood level and containment radiation level is below the post-accident radiation alarm setpoint.

Basis of Status Tree Branches

Branch Description: Containment Pressure Less Than 46 PSIG

If containment pressure is greater than design pressure, an extreme challenge to the containment barrier exists. The challenge does not necessarily come from the pressure alone, but rather from the potential pressure spike which could result from a hydrogen ignition. The total pressure could then potentially exceed the strength of containment. Also, above containment design pressure, leakage may exceed design basis limits. It is expected that containment pressure suppression equipment should be able to maintain pressure below design pressure. If not, then operator action is necessary to check containment functions and a RED priority is warranted. The appropriate procedure for function restoration is 1FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.

Branch Description: Containment Pressure Less Than 23 PSIG.

At a pressure below design pressure, it is unlikely that even a hydrogen ignition could result in sufficient overpressure to fail containment. Pressure above 46 psig indicates a significant energy release to containment and merits prompt operator action to ensure operation of containment pressure suppression equipment and performance of containment isolation. Pressure above 23 psig requires ensuring main steamline isolation and is considered a severe challenge to the containment barrier and an ORANGE priority is warranted. The appropriate guideline for function restoration is 1FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.

Branch Description: Containment Sump Level Less Than 8.0 FT..

High energy line breaks could result in a large volume of water being pumped into containment. As the water level rises, it might threaten the availability of equipment required for long term cooling of the core and/or containment. Such a high water level is considered a severe challenge to the containment barrier and an ORANGE priority is warranted. The appropriate procedure for function restoration is 1FR-Z.2, RESPONSE TO HIGH SUMP B LEVEL.

<u>Branch Description</u>: Containment Radiation Less Than 200R/HR

Normally, containment building radiation levels are fairly low and constant. However, during an accident, significant radioactivity may be released into the containment atmosphere. In-containment systems are available to filter and scrub the contaminants from the atmosphere, and radiation alone does not represent a threat to containment integrity. This is considered a not satisfied condition and a YELLOW priority is warranted. The appropriate procedure for function restoration is 1FR-Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION. If containment radiation is less than 200R/Hr (Radiation Alarm Setpoint for the Post Accident Rad. Monitor), then the function is satisfied.

Question : RO 54

Comment:

New police sirens sound more like our fire alarm – they are shorter in cycle. A & B should be correct since it is too subjective based on the description of the siren.

Facility Response:

The cycle of the evacuation alarm is significantly longer than the fire alarm and should be recognizable by the information given.

Additionally, the action for distracter B is not correct. Containment boundary control is maintained by tagging closed the redundant valves while performing the post-LLRT lineup. Continuing the lineup is not expected provided the job is in a safe condition.

Recommend no change to this question.

Level	RO	Tier	2	Group	1	K/A#	103 A2.04	Imp. RO	3.5*	Imp. SRO

o4. You are an extra operator performing a post-LLRT lineup on a containment penetration.

You hear a variable tone siren (wailing, like a police siren) with about a 4-second cycle, but due to noise in the area are unable to hear the announcement that follows.

What has occurred, and what action is required?

a. Containment Evacuation Alarm. Immediately evacuate containment using the nearest airlock and card out of containment.

b. Fire Alarm. Complete the valve lineup to ensure the containment penetration is isolated, then exit containment and assist the Fire Brigade or Control Room as directed.

c. High Flux at Shutdown Alarm. Immediately evacuate containment using the nearest airlock and card out of containment.

d. Site Evacuation Alarm. Complete the valve lineup to ensure the containment penetration is isolated, then exit containment and report to the North Warehouse.

ANSWER: A

planation: **A: Correct, the Evacuation Alarm is a variable tone alarm. Per the Outage Handbook, all plant personnel are required to evacuate containment immediately via the nearest airlock. Containment closure is maintained since the LLRT maintains one valve closed in the penetration at all times.**
B: Incorrect, the fire alarm is a variable tone but on a 1-second cycle, more of a "whoop whoop" sound. Action is correct per F5.
C: Incorrect, the HFASD alarm is a continuous tone alarm per the Outage Handbook. Action is correct.
D: Incorrect, plausible as the Containment Evacuation and Site Evacuation alarms sound the same. However, the action described does not meet the requirements of 'a' since it is not an immediate evacuation, and operators are to report to the OSC.

Technical References: **NMC Outage Handbook**
F5 Firefighting section 3.5

Objective: **NGA01L001H Emergency Response/Preparedness #3**

KA Statement: **Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.**
Containment evacuation (including recognition of the alarm)

Cognitive Level: 1-P **10CFR55.41 10** **10CFR55.43** **New Question No**
Bank ID: Point Beach **Question ID: 103 A2.04** **Modified: Yes** **Last NRC Exam: 9/03**

alarm is received. If you are hearing impaired, contact RP for a dose alarm light for your ED. Listen to your dosimeter. It will remain quiet unless there is something wrong.

ALARMS:	All alarms **SHALL** be reported to an RPS.
DOSE RATE ALARM:	Continuous beeping. Exit the area and contact the RPS. Dose rates may have *changed*.
DOSE ALARM:	On and off tone. Your dose limit for the RWP has been reached. Exit the area and contact the RPS. On and off tones may also indicate a dosimeter malfuntion which is another good reason to get out of the area.

CONTAINMENT ENTRY

Containment entry is coordinated through the Access Facility.

CONTAINMENT EVACUATION

A Containment evacuation is required for fuel handling accidents, unexpected criticality, rapidly decreasing refueling water level, or at the discretion of the Control Room and Refueling SRO (Senior Reactor Operator).

The Containment Evacuation Alarm is a variable tone alarm and the High Flux at Shutdown Alarm is a continuous non-variable tone. Both alarms require all plant personnel to evacuate the containment by the nearest airlock.

When the alarm occurs, the Control Room will make an announcement on the Plant PA System. All personnel in the SFP and affected unit's annulus and containment will evacuate by the nearest airlock to the dress-out area (basketball court) and card out of containment. This will allow for accountability of personnel. During the evacuation, Radiation Protection personnel will make a rapid search of the containment to ensure all personnel have responded to the alarm. If necessary, a search and rescue team will be implemented.

AVOID THESE COMMON INDUSTRY PITFALLS

- High Rad Area violations.
- Moving radiological boundaries.
- Reaching across boundaries.
- Throwing contaminated protective clothing into hampers.
- Ignoring Electronic Dosimeter Alarms.
- Entering RCA without ED or ED not turned on.

NMC Outage Handbook 2R23

The Brigade Chief **SHALL** direct the activities of the Fire Brigade, keep the CR informed and direct the securing of equipment. He **SHALL** direct the guardhouse to escort the local fire department personnel to a particular plant entrance and ensure that a runner is sent to this entrance. He **SHALL** direct offsite fire department personnel on any actions involving plant safety. If the plant fire brigade is able to extinguish the fire before the local fire department arrives, the fire department **SHALL** be so informed. However, the fire department personnel and vehicle **SHALL** be escorted to the scene of the fire in the same manner as if the emergency still existed. This is necessary to expedite the investigation that is required whenever the fire department responds to a call.

The Brigade Chief **SHALL** be ultimately responsible for firefighter safety and accountability. He/She may designate an individual to assist in this function if available, however, the Brigade Chief retains the ultimate authority on issues regarding the safety of the firefighters.

3.3 Assistant Chief

An APEO from the Unit 2 Turbine Building or Auxiliary Building, **SHALL** fulfill the duties of the Brigade Chief in the absence of the Brigade Chief.

3.4 Fire Fighters

BOP operator assigned to the affected building **SHALL** answer Fire Alarm page immediately, proceed to the fire zone in alarm, report findings to CR, return to scene of fire, attempt to extinguish it if this can be accomplished without undue personnel hazard. Report to Brigade Chief when he arrives, then leave scene to don fire fighting clothing/equipment and return. BOP operators assigned to the unaffected building **SHALL** don fire fighting clothing and Self-Contained Breathing Apparatuses (SCBA's) at Fire Brigade changeout area and then report to the scene of the fire bringing additional SCBA's, protective clothing, and other specialized equipment as needed.

3.5 Runners

This position is filled from those operations personnel available for, but not required for, fire fighting. Activities include:

3.5.1 Assisting the Brigade Chief with firefighter safety/accountability.

Question : RO 67

Comment:

F5 Appendix B does not list alternative methods in order of preference. It only gives a list of alternative methods and leaves it up to the operator.

Facility Response:

F5 Appendix B addresses five methods of communications:

- Radio (directed to obtain prior to evacuation)
- Telephone (phone numbers for stations have been added to the attachments)
- Gaitronics paging
- Sound Powered Telephone
- Runners

F5 Appendix B Assumption 2.1.5 states that runners are the only credited means of communications. The other communication methods have connections in the control or relay room BUT they may be used only as long as they continue working. This guidance is repeated in section 3.0 and the note at the top of section 4.0.

The answer given was based on the procedural note for the phone number (listed at the top of attachment A), but each method has unique advantages and disadvantages, and one or more may be used to establish and maintain communication. As an example, runners are specifically credited and not affected by the fire. Sound powered phones can be worn on station and would connect all the stations together. Telephones would be the fastest direct form of communication if available. Paging systems may be used to get the attention of other personnel and direct them to establish communications.

Since the communications methods in the question are not listed in order of preference, any of the answers given could be considered "preferred".

Recommend deleting this question from the exam.

Level	RO	Tier	G	Group	K/A#	2.1.16	Imp. RO	2.9	Imp. SRO

67. The control room has been evacuated per F5 Appendix B CONTROL ROOM EVACUATION-FIRE. You are conducting Attachment D, Unit 2 Reactor Operator Actions.

You have been unsuccessful in contacting the Shift Supervisor at the Hot Shutdown Panel using the radio.

What is the preferred alternate method of communication with the Shift Supervisor at the Hot Shutdown Panel?

a. Have an extra person (runner) deliver a written message

b. Sound Powered Telephone on Circuit 1

c. Gaitronics Page

d. Telephone

ANSWER: D

Explanation: A; Incorrect, this is slowest communication form and would only be used if fire takes out the radios and telephone. It is credited per Assumption 2.1.5 as the other methods may be affected by the fire.
B: Incorrect, can be used but no direction is given to monitor this channel and there is no way to signal the SS that you are on this circuit. Also, Circuit 1 is not the emergency circuit.
C: Incorrect, would work to get his attention if away from the HSD panel area, but communication using this method would be distracting to the other personnel in the plant.
D: Correct, the HSD panel phone number is listed as a note at the top of the attachment A. Per Assumption 2.1.5, as long as the phone continues to work it may be used. Phone use is preferred over plant paging (may interfere with other announcements) and runner (slowest). There is no procedural direction for the SS to be monitoring the sound powered phone.

Technical References: F5 App B

Objective: P8140L-229 A.7

KA Statement: Ability to operate plant phone, paging system, and two-way radio.

Cognitive Level:	1-P	10CFR55.41	10	10CFR55.43		New Question	No
Bank ID:	Farley 1	Question ID:	2.1.16	Modified:	Yes	Last NRC Exam:	5/03

1.0 PURPOSE

This procedure provides the steps necessary to achieve and maintain Mode 3, Hot Standby and to cooldown to Mode 5, Cold Shutdown in the event of a catastrophic fire that results in the functional loss and/or evacuation of the Control Room/Relay Room.

2.0 GENERAL DISCUSSION

2.1 Assumptions

2.1.1 From the Control Room the reactors and turbines are tripped, the RCPs are tripped, and the MFW Pumps are tripped, and the MSIVs and PORV block valves are CLOSED. Additionally, the Containment Spray Pumps are tripped. Control switches for the listed pump motors are placed in the "PULL-TO-LOCK" position as a defense-in-depth measure to preclude spurious start. Also, the Pressurizer Spray Valve control transfer switches are placed in "MANUAL" to mitigate spurious actuation.

2.1.2 Outside the Control Room, removing power for the RCS PORVs and isolating steam downstream of the MSIVs provides assurances that the manual actions taken before leaving the Control Room will not be negated by subsequent spurious actuation signals resulting from the postulated fire.

2.1.3 The procedures assume normal letdown and charging were in service prior to the fire.

2.1.4 No credit is taken for manual action from the Control Room to control pressurizer pressure or level.

2.1.5 The sound powered headphone system and Gai-tronics page system have termination boxes in the Relay Room. The radio system power supply cabling passes through the Relay Room also. These communications systems may be affected by the fire. As long as they continue to work, they may be used. However, the only credited communication is face-to-face.

3.0 INITIAL EVACUATION ACTIVITIES

The Unit 1 Shift Supervisor is tasked with determining the continued habitability of the Control Room if a fire occurs in the Control Room or Relay Room. Once the decision is made to evacuate the Control Room, specific action must be taken to place the units in a stable, Mode 3, Hot Standby condition. Section 3.0 of this procedure describes these activities and the specific actions appear in the form of Attachments.

Attachment A through Attachment H describe the individual activities, based on assigned position at the time of the evacuation. Each person is responsible for obtaining their designated Attachment and performing the described actions. These Attachments are provided at designated locations and are readily available to the assigned personnel.

Normal plant communication systems may be affected by the fire. If available, they can be used. IF unavailable, THEN use a runner to provide face-to-face communication.

Once the necessary actions have been completed to place the units in Mode 3, Hot Standby, a status evaluation will be performed by the Shift Manager, Shift Supervisors and TSC personnel. A controlled cooldown to Mode 5, Cold Shutdown is described in Section 4.0 as a guideline for cooldown activities. Depending on actual conditions, alternative strategies may be employed as directed by plant management.

The following are the specific responsibilities of On-Shift Operations personnel upon initial evacuation:

3.1 Unit 1 Shift Supervisor

Upon making the decision to evacuate the Control Room, the Unit 1 Shift Supervisor will announce the evacuation, assure appropriate notifications are made, determine if SCBA use is required, and then proceed to the Hot Shutdown Panels in the Auxiliary Feedwater Pump Rooms. Using Attachment A as a guide, he will then align equipment for operation from the Hot Shutdown Panels and locally in the AFW Pump Room per Attachment I. With face-to-face instructions for the various operators performing activities throughout the plant, he will direct additional activities as necessary.

4.0 SUBSEQUENT ACTIONS

NOTE	**Normal plant communication systems may be affected by the fire. If available, they can be used. If unavailable, then use a runner to provide face-to-face communication.**

The following instructions provide the necessary actions to cooldown the units from Mode 3, Hot Standby conditions achieved in Section 3.0.

4.1 **Rack** breakers to the "DISCONNECT" position for the SI Pumps and CS Pumps. ______

4.2 **Check** CC System status:

4.2.1 **Contact** SM to check status of CC Systems using ERCS screen "CC1" for each unit. ______

OR

4.2.2 **Contact** Auxiliary Building Operators to check local indications. ______

4.2.3 **Check** at least one CC Pump running for each unit. IF NOT, THEN:

A. **Stop** any RCP that is running without CC flow. ______

B. **Verify** DC Control Power has been restored to applicable 4KV Safeguards Bus. ______

C. Locally **start** desired Component Cooling Pump as follows:

1. **Depress** local **RESET** pushbutton. ______

2. **Depress** local **START** pushbutton. ______

4.2.4 **OPEN** the following MCC breakers:

MCC 1K1-B3, 11 CC HX CLG WTR MV-32145 ______

MCC 2K1-B3, 21 CC HX CLG WTR INLET MV-32160 ______

Question : RO 71

Comment:

Procedurally, the earliest in-service purge may be lined up is in Mode 5, but the earliest by Tech Specs is Mode 3. Nothing prevents changing our procedures to perform leak testing to initiate in-service purge earlier in Mode 3. Recommend accepting "A" and "C".

Facility Response:

Facility operating procedures do not allow operation of Containment In-Service Purge until MODE 5 is reached. This action is required as the blank flanges must be removed and the inside containment dampers do not meet containment integrity requirements. **Recommend no change to this question.**

'.evel RO Tier G Group K/A# 2.3.9 Imp. RO 2.5 Imp. SRO

71. You are the RO during a shutdown for a refueling outage. The Containment HP requests that Containment In-Service Purge be placed in service "as soon as possible" to reduce dose to workers.

Which of the following is the EARLIEST plant mode reached that will allow for establishment of Containment In-service Purge?

a. Entry into MODE 3 Hot Standby.

b. Entry into MODE 4 Hot Shutdown.

c. Entry into MODE 5 Cold Shutdown.

d. Entry into MODE 6 Refueling.

ANSWER: C

Explanation: A: Incorrect, requires removal of blank flanges and installation of spoolpiece, which cannot be done while Containment Integrity is required.
B: Incorrect, Containment Integrity still required.
C: Correct, in MODE 5 containment integrity is not required and purge will be placed in service.
D: Incorrect, not the earliest plant mode to place it in service.

ıechnical References: 1C19.2
Fig B19-9
C1.1.19-1 penetration 42B, 43A

Objective: P8180L-009E, Objective 6

KA Statement: Knowledge of the process for performing a containment purge.

Cognitive Level: 1-P **10CFR55.41** 10 **10CFR55.43** **New Question** Yes
Bank ID: **Question ID:** **Modified:** **Last NRC Exam:**

1.0 PURPOSE

This procedure describes the operations necessary for sub-systems associated with the containment vessel, to maintain suitable environments for structures, components and personnel. Also included is the operation of special ventilation systems which surround the containment vessel penetrations and which maintain these areas in a vacuum following specific events.

2.0 PREREQUISITES

2.1 IF a short-term containment release has not occurred since the unit was placed in Mode 5, Cold Shutdown, THEN a Containment Pre-Release Authorization **SHALL** be completed and approved by the Duty Chemist.

2.2 **Containment Purge System:** A minimum of 2 channels of Radiation Monitors (i.e., **1R-12** and **1R-22**) **SHALL** be operable and the associated safeguards racks **SHALL** be energized.

2.3 **Containment In-Service Purge System:** A minimum of 2 channels of Radiation Monitors (i.e., **1R-12** and **1R-22**) **SHALL** be operable and the associated safeguards racks **SHALL** be energized.

2.4 Containment Purge or In-Service Purge supply and exhaust blank flanges have been removed in accordance with D61.

3.0 PRECAUTIONS

3.1 IF containment pressure exceeds ≈1.3 psig, on ERCS computer point 1U1000A, THEN contact Rad Protection to obtain a containment sample for a depressurization per C19.4, Post LOCA Vent System Depressurizing the Containment Vessel.

3.2 Both the Containment Purge System and the Containment In-Service Purge System are limited to 0.5 psig. IF the initial containment pressure exceeds 0.5 psig, THEN prior to placing either purge in service, the pressure **SHALL** be reduced to less than 0.5 psig in accordance with:

- C19.4, Post LOCA Vent System Depressurizing the Containment Vessel

  OR

- C11, Radiation Monitoring System, Steps for controlling containment pressure thru Sample Outlet Test Line

4.0 SPECIAL EQUIPMENT AND PERSONNEL REQUIRED

4.1 Calibrated Torque Wrench

4.2 The wrench/socket sizes required are given in the table below:

Penetration	Size
Station Air 1(2) 19	1 1/16" & 1 1/4"
Demin Water 1(2) 49B(55)	1 1/16" & 1 1/4"
Fire Protection 1(2) 27B(51)	1 1/16" & 1 7/16"
Inservice Purge 1(2) 42B and 43A (53 and 52)	1 13/16" 1 11/16"
Cont Purge 1(2) 25A and 25B	2 3/8"

(Socket, Extension, Swivel, and Wrenches)

4.3 Rigging will be necessary for the inservice purge spool pieces removal procedure.

4.4 Air impacts may be used for removing the containment purge flanges.

4.5 IF "O-rings" or gaskets are to be replaced in the spool piece removal procedure, THEN they **SHALL** be of radiation qualified material and documented on the Work Order tracing them to the P.O. number.

5.0 SPECIAL CONSIDERATIONS

5.1 The integrity requirements of C19.9, "Containment Boundary Control During Mode 5, Cole Shutdown and Mode 5, Refueling **SHALL** be observed.

5.2 T.S.3.6.1 requires the RCS to be in MODE 5, Cold Shutdown for this work.

5.3 T.S.3.7.12 and Basis allows openings in the ABSVZ as long as they are logged per D54 and can be reduced to less than 10 sq. ft. within 6 minutes following an accident. The SA, DE and FP instructions within this procedure create temporary openings until they are sealed for the outage with plywood or permanently resealed.

5.4 H4, ODCM requires SP 1263[2263] to be performed before use of Big Purge per Attachment I[J].

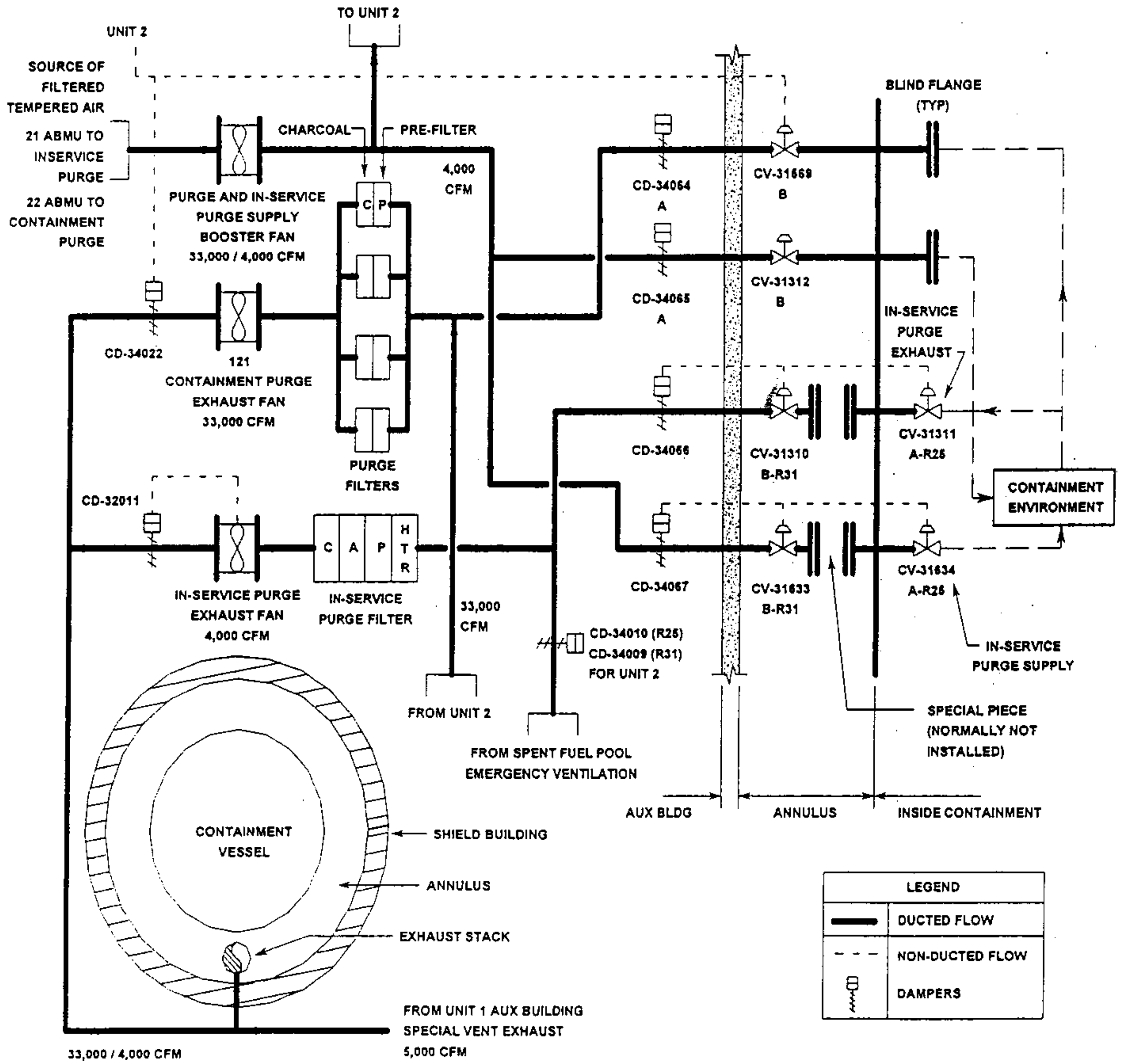


FIGURE B19-9 - CONTAINMENT PURGE AND INSERVICE PURGE SYSTEM

CHECKLIST	CONTAINMENT AND SHIELD BUILDING PENETRATION CHECKLIST - UNIT 1	NUMBER: C1.1.19-1
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LOCATION: Annulus (Top Elevations Down)

PEN#	COMPONENT	DESCRIPTION	STATUS	INITIAL
42B	BLIND FLANGE	INSERVICE PURGE SPLY BLIND FLANGE IN ANNULUS ON CNTMT PEN.	INSTALLED	
	TEST CONNECTIONS	TEST PLUGS (2) ON BLIND FLANGE	SWAGELOK CAPS	
	MANWAY	INSERVICE PURGE SUPPLY MANWAY	INSTALLED	
	CV-31633	INSERVICE PURGE SUPPLY	INTACT/ AIR OPEN	
43A	BLIND FLANGE	INSERVICE PURGE EXHAUST BLIND FLANGE IN ANNULUS ON CNTMT PEN.	INSTALLED	
	TEST CONNECTIONS (2)	TEST PLUGS (2) ON BLIND FLANGE	SWAGELOK CAPS	
	MANWAY	INSERVICE PURGE EXHAUST MANWAY	INSTALLED	
	CV-31310	INSERVICE PURGE EXHAUST	INTACT/ AIR OPEN	
MAINT AIRLOCK	FLANGE	MAINTENANCE AIRLOCK FLANGE	INSTALLED	
	TEST PLUG	TEST PLUG ON FLANGE	SWAGELOK CAP	
	DOOR SEAL TEST VALVES	MAINTENANCE AIRLOCK DOOR SEAL TEST VALVES	INTACT	
	INTERLOCK	MAINTENANCE AIRLOCK INTERLOCK	IN-SERVICE	