QID:	0037	**Rev:**	002	**Rev Date:**	10/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	NRC Exam Bank	**Originator:**	Hatman		
10CFR55_41:	41.10	**10CFR55_43:**	43.3	**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.11						
RO Tier:	3	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**	3	**SRO Group:**	2	**SRO Imp:**	3.4
Description:	Knowledge of the process for controlling temporary changes.										

Question:

Which ONE (1) of the following tasks should be classified as a POTENTIAL TEMPORARY ALTERATION? (Reference material provided)

A. Performing a channel calibration in which the procedure requires installing jumpers to electrically bypass automatic actuation.

B. A blank flange is installed on a non-safety tagged out line while rerouting the line under an approved MAI.

C. Connecting cables from 480V Motor Control Center (MCC) to a temporary power panel for outage maintenance support.

D. Maintenance technicians installing a temporary drain hose to support changing oil in a tagged out pump.

Answer:

C. Connecting cables from 480V Motor Control Center (MCC) to a temporary power panel for outage maintenance support.

Notes:

Answer "A" is incorrect since it is covered by an approved procedure.
Answer "B" is incorrect since it is covered under a tagout, then the system is Out of Service.
Answer "D" is incorrect since it is covered under an approved MAI and tagged out.

Provide Control Of Temporary Alteration Procedure 1000.028, Rev 023-01-0, Attachment 1 (Page 34) as a reference on closed exams

References:

1000.028, Control of Temporary Alterations, Rev 023-01-0, Steps 6.3, and 6.6 and Att. 1 and 2
ANO-S-LP-RO-ADMIN, Revision 1, Objective 4.0
ANO-S-LP-SR0-ADMIN, Revision 4, Objective 4.0

Historical Comments:

Rev 001 - 8/11/98 - Added "Reference material provided" to stem and added note that pages 26 and 27 of 1000.028, Rev 20 will be provide as reference material with exam per NRC review comment.
Rev 002 - 09/28/2001- Modified stem to reflect task screening to determine if task is a Potential Temp Alt per Attachment 1. Added non-safety and changed Job Order to MAI in distracter B, added tagged out in distracter

D. Updated reference revisions and changed page numbers to steps/attachments. This question was used on the 1998 RO and SRO exam.

Questions For 2002 SRO/RO Exam

17-Jan-02

QID: 0070 **Rev:** 002 **Rev Date:** 01/10/2002 **RO Select:** No **SRO Select:** Yes **Points:** 1.00

Lic Level: S **Difficulty:** 3 **Taxonomy:** AP **Source:** NRC Exam Bank **Originator:** Coble

10CFR55_41: 41.5 & 41.10 **10CFR55_43:** **Section:** 4.1 **Type:** Generic EPE

System Title: Anticipated Transient Without Scram (ATWS **System Number** 029 **K/A:** EK3.12

RO Tier: **RO Group:** **RO Imp:** **SRO Tier:** 1 **SRO Group:** 1 **SRO Imp:** 4.7

Description: Knowledge of the reasons for actions contained in EOP for ATWS.

Question:

If Reactor power is not lowering after a plant trip, Standard Post Trip Actions (SPTAs), 2202.001, Contingency Step 3.A.2 directs the operator to open the feeder breakers to 480 VAC Load Center busses 2B7 and 2B8 for ten (10) seconds and then reclose them.

The reason for re-closing these breakers is to:

A. Restore power to the CEDMCS in order to verify that all CEAs have fully inserted.

B. Re-energize the 2B7 and 2B8 busses before the under voltage relays strip the individual loads from the busses.

C. Restore power to the Pressurizer backup heaters for proper pressure control.

D. Restore power to Component Cooling Water Pump 2P33C and 480V MCCs.

Answer:

D. Restore power to Component Cooling Water Pump 2P33C and 480V MCCs.

Notes:

CEDMCS control power is not interrupted. Load centers do not have under voltage relays to strip loads, and backup heaters are powered from Load Centers 2B9 and 2B10.

This question was generated from a randomly selected K/A to be part of the SRO exam and not on the RO exam; however, this question is not one of the 25 10 CFR 55.43 category questions selected for this exam. Four additional questions were selected to be on the SRO exam that are not on the RO exam to in order to comply with the NUREG 1021 guidance to have a balance of K&A selections on the initial sample plan. One of these 4 happen to fall into the 10 CFR 43 category so there are actually 26 SRO only questions on the SRO exam that are in the 10 CFR 43 category.

References:

2202.001, Revision 005-00-0, Step 3.A.2 (Standard Post Trip Actions)
2107.001, Rev 44-06-0, Page 31 (Electrical System Operation-Power to CCW Pump C)
ANO-2-LP-RO-ESPTA Objective 8

Historical Comments:

Rev 001 - 08/11/98 - Corrected procedure step in stem from "2.2" to "2.A.2" per NRC review comment.

Revision 002 - 10/29/2001 - Updated References and added "if Reactor Power is not lowering after a plant trip" to the stem. This question was used on the 1998 SRO only exam. 1/10/2002. Added a clarification note as to why this SRO only question used on the 2002 exam is not from the 10 CFR 55.43 criteria based on NRC question. BNC

QID:	0136	**Rev:**	002	**Rev Date:**	10/15/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	NRC Exam Bank	**Originator:**	Hatman		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	High Reactor Coolant Activity	**System Number**	076	**K/A:**	AA2.02						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.4
Description:	Ability to determine/interpret corrective actions required for high fission product activity in RCS.										

Question:

Given the following plant conditions:

Unit is operating at 100% power.
E-bar is 5.3 microcuries/gram.
RCS sample shows specific activity of 25 microcuries/gram.

Which one of the following states the required action(s) for this condition?

A. Restore RCS activity to within limits within 1 hour or be in at least hot standby in the following six (6) hours.

B. Be in at least HOT STANDBY with T-ave less than 500°F within six (6) hours.

C. No actions required due to LCO conditions being satisfied.

D. Perform sampling of RCS activity within the next six (6) hours.

Answer:

B. Be in at least HOT STANDBY with T-ave less than 500°F within 6 hours.

Notes:

Tech Spec specific activity limit is less than 100/E-bar = 100/5.3 = 18.87 microcuries/gram. Actual specific activity is 25 microcuries/gram so Tech Spec 3.4.8 action applies.

References:

Tech Spec 3.4.8, Amendment 92
2203.020, Rev 007-05-0, High Activity in RCS, Step 4
ANO-2-LP-SRO-TS,Rev 07, Technical Specifications, Objectives 3.0, 4.0
ANO-2-LP-SRO-EAOP, Rev 07, Abnormal Operating Procedures, Objective 23.0

Historical Comments:

Rev 001 - 08/11/98 - Revised distracter "D" from "Perform sampling and analysis per Table 4.4-4 until RCS activity limits are satisfied" with "Perform sampling of RCS activity within the next six (6) hours" to eliminate the possibility of distracter "D" being considered a correct answer.
09/10/01 - Used on 1998 SRO Written Exam. No modifications made.
10/16/01 - Rev 002 - Added AOP and LP Obj to references.

QID:	0161	**Rev:**	001	**Rev Date:**	10/15/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	An	**Source:**	NRC Exam Bank	**Originator:**	Hatman		
10CFR55_41:		**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K/A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.10						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	3	**SRO Imp:**	3.3
Description:	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.										

Question:

Given the following plant conditions:

* A small break Loss of Coolant Accident is in progress.
* RCS pressure is 1600 psia.
* Containment pressure is 16 psia and slowly increasing.
* Containment Hi Range Radiation Monitors read 13 R/Hr and increasing.
* Containment Low Range Radiation Monitors are in alarm and trending up
* No operator actions have been performed.
* All systems function as designed

Which one (1) of the following actions should be performed?

A. Isolate letdown.

B. Manually actuate SIAS.

C. Manually actuate CIAS.

D. Manually actuate CSAS.

Answer:

C. Manually actuate CIAS.

Notes:

SIAS would already have automatically actuated, therefore L/D would already have isolated. CIAS should be manually actuated due to high Containment Radiation to preclude the possibility of releases from containment. CSAS should not be automatically or manually actuated during a Small Break LOCA at the existing containment pressure in order to maintain RCPs running

References:

2203.012J, Revision 028-04-0, 2K10-A6, (Alarm 2K10 Corrective Action for CNTMT Radiation HI)
ANO-2-LP-RO-ELOCA, Revision 05, Objective 6.0

Historical Comments:

09/28/2001 - Rev 001 - Updated reference revisions, added low range radiation monitors in the conditions due to information note added to Alarm Corrective Action concerning validating high range radiation monitors

with the low range radiation monitors.
10/16/01 - This question was used on the 1998 exam.

QID:	0219	**Rev:**	001	**Rev Date:**	10/16/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	NRC Exam Bank	**Originator:**	Hatman		
10CFR55_41:	41.2 to 41.9	**10CFR55_43:**		**Section:**	3.8	**Type:**	Plant Service System				
System Title:	Containment Purge System	**System Number**	029	**K/A:**	K1.03						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.6	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.8
Description:	Knowledge of the physical connections and/or cause-effect relationship between the Containment Purge System and Engineered Safeguards.										

Question:

Containment Purge will be automatically secured on all of the following EXCEPT:

A. Hi-Hi alarm signal on Radwaste Area Disch 2VEF-8A/B Process Rad Monitor 2RITS-8542.

B. Hi-Hi alarm signal on Containment Purge Disch 2VEF-15 Process Rad Monitor 2RITS-8233.

C. Hi Containment Pressure of 18.3 psia.

D. Pressurizer Pressure of 420 psia with a variable low pressure setpoint of 450 psia.

Answer:

A. Hi-Hi alarm signal on Radwaste Area Disch 2VEF-8A/B Process Rad Monitor 2RITS-8542.

Notes:

Answers B, C and D are wrong because these signals will cause an automatic termination of the containment purge system. All of these three signals will cause at least one supply and return containment Isolation to automatically close. B because of high exhaust activity, C because of a SIAS and CIAS, and D because of a SIAS signal. The isolation closing will cause a low suction pressure trip of the exhaust unit and the supply unit has to see the exhaust unit running or it will trip after a 10 second time delay.

References:

ANO-2-LP-RO-CVENT, Revision 0, Objective 13
STM 2-9, Containment Cooling and Purge Systems, Rev 6, Sections 7.3, 7.5, 7.6.
STM 2-62, Radiation Monitoring System, Rev 6, Sections 2.2.7.2, 2.2.7.3
OP 2104.033, Revision 040-06-0, Section 3.0

Historical Comments:

10/16/2001 - Rev 001 - Updated reference revisions and change source to "ANO NRC Exam Bank" because this question was used on both the 2000 RO and SRO exams.

QID:	0296	**Rev:**	000	**Rev Date:**	03/10/2000	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	NRC Exam Bank	**Originator:**	Woolf		
10CFR55_41:	41.12	**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K/A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.1						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	3	**SRO Imp:**	3.0
Description:	Knowledge of 10CFR20 and related facility radiation control requirements.										

Question:

Given the following plant conditions:

* 100% steady state Power.
* RCS Leakrate Calculation has increased from 0.11 gpm to 0.65 gpm over past two (2) hours.
* RCP 2P32A Lower Seal Pressure (2PI-6004) has decreased from 1520 psia to 685 psia.
* A Containment Building Power entry is being planned to inspect 2P32A seal area and attempt to locate and isolate source of increased RCS leakage.

Which of the following statements reflect the approval(s) required for this power entry by Form 1601.300A, Reactor Building Power Entry Checkoff List?

A. The Unit 2 Plant Manager AND the Manager, Radiation Protection.

B. The Manager, Unit 2 Operations AND any Superintendent, Radiation Protection.

C. The Unit 2 Shift Manager OR the Manager, Radiation Protection.

D. The Unit 2 Shift Manager OR the Superintendent, Radiation Protection.

Answer:

A. The Unit 2 Plant Manager AND the Manager, Radiation Protection.

Notes:

To check RCP Seal Area entry must be made inside the Bio-shield. This requires Plant Manager and Manager, Rad Protection. Other distracters are valid for different areas.

References:

1601.300, Job Coverage, Rev 008-00-0, Attachment 3, Step 5.1.4.B.3
1601.300, Reactor Building Power Entry Check-off List, Rev. 008-00-0
ANO-2-LP-SRO-CFR, Federal Regulations, Rev 04, Objective 4

Historical Comments:

03/10/00 - Rev 000 - Developed to replace QID 0259 on 2000 SRO Exam.

QID:	0298	**Rev:**	002	**Rev Date:**	12/06/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	NRC Exam Bank	**Originator:**	Woolf		
10CFR55_41:	41.12	**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K/A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.10						
RO Tier:	3	**RO Group:**	3	**RO Imp:**	2.9	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.										

Question:

Which of the following statements describes Unit 2 Control Room Watchstander (SM, CRS, SE, CBOR and CBOT) respirator qualification requirements?

A. Self contained breathing apparatus (SCBA), due to control room watch standing requirements.

B. Self contained breathing apparatus (SCBA), due to fire brigade watch standing requirements.

C. Air purifying respirators, due to fire brigade watch standing requirements.

D. Air purifying respirators, due to control room watch standing requirements.

Answer:

A. Self contained breathing apparatus (SCBA), due to control room watch standing requirements.

Notes:

Self contained breathing apparatus are required due to the inability of the purifying respirators to filter out the toxic gasses that could make its way into the control room in case of a chemical release or fire. This makes answers C and D wrong. The control room staff does not have any responsibility to fight fires with the fire brigade, only to direct communications and maintain safety of the plant so answer B is wrong.

References:

1015.001, Conduct of Operations, Rev 053-02-0, Section 3.3.13, Section 6.0
LER 97-02, Operators not having SCBA glasses and respirator qualifications.
CR-ANO-C-1997-0145, Operators not have SCBA glasses and respirator qualifications.
ANO-S-LP-RO-ADMIN, Admin Procedures, Rev 01, Objective 5.

Historical Comments:

03/12/00 - Rev 000 - Developed for 2000 RO Exam to replace QID 0239.
04/12/00 - Rev 001 - Deleted "best" from stem based on NRC examiner comments.
10/15/00 - Rev 002 - Updated references. Changed SS to SM in stem.

QID:	0300	**Rev:**	001	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	AP	**Source:**	NRC Exam Bank	**Originator:**	Woolf		
10CFR55_41:	41.5	**10CFR55_43:**	43.5	**Section:**	3.8	**Type:**	Plant Service Systems				
System Title:	Spent Fuel Pool Cooling System	**System Number**	033	**K/A:**	A2.03						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.1	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.5

Description: Ability to (a) predict the impacts of abnormal spent fuel pool water level or loss of water level on SFPC System; (b) based on those predictions use procedures to correct, control or mitigate the consequences of those malfunctions or operations.

Question:

Given the following plant conditions:

* Mode 6 operation with core offload in progress.
* A complete Loss of Offsite Power occurs.
* Diesel Generator 2DG1 is supplying electrical bus 2A3 and 2DG2 failed on startup.
* Alternate AC (AAC) Diesel Generator is NOT available due to maintenance on governor.
* Annunciator 2K11-K5 "Fuel Pool Temp High" is actuated.

For the above plant conditions, which of the following would be an available source of makeup for the SF Pool, if makeup is required ?

A. Loop 1 Service Water using Service Water Pump 2P4A.

B. Refueling Water Tank using Spent Fuel Pool Purification Pump 2P66.

C. Boric Acid Makeup System using Boric Acid Makeup Tank Pump 2P39A/B and Reactor Makeup Water Pump 2P109A/B.

D. Boron Management System Holdup Tanks using Holdup Tank Recirc Pump 2P48

Answer:

A. Loop 1 Service Water using Service Water Pump 2P4A.

Notes:

A is correct because #1 EDG is running supplying 2A3 bus which powers SW Pump 2P4A.
B is incorrect because 2P66 is power from Non-Vital AC MCC 21.
C is incorrect because 2P39A/B supplied by Green Vital power/2P109A/B supplied by Non-Vital AC.
D is incorrect because Holdup Tank Recirc Pump 2P48 is powered from Non-Vital MCC 2B31

References:

ANO-2-LP-RO-SFP, Revision 0, Objectives 5,6,and 7
2107.001, Electrical System Operations, Rev 044-06-0, Attachments D & I.
2107.002, ESF Electrical System Operations, Rev 015-01-0, Attachment D.
2104.006, Fuel Pool Systems, Rev 018-05-0, Sections 10, 11, 12 & 14.

Historical Comments:

03/13/00 - New question added for 2000 SRO Exam to replace QID0197.
10/16/2001 - Rev 001 - Updated Reference Revision Numbers. This question was used on the 2000 SRO exam. 1/17/2002- Spelled out all acronyms based on NRC feedback

QID:	0308	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	RCP MALFUNCTIONS	**System Number**	015	**K/A:**	AA1.06						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.1	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	2.9
Description:	Ability to operate and or monitor the CCWS as it applies to a Reactor Coolant Pump Malfunction.										

Question:

The plant is at 100% power.

* CCW Pump 2P33C is operating on CCW Loop II.
* Annunciators 2K11-A1/A3/A5/A7, RCP CCW DISCH FLOW LO actuate.
* The CBOT notes that CCW Supply to RCP Isolation, 2CV-5236-1, indicates closed.

The operating crew completes actions to restore CCW to RCPs in accordance with 2203.025, RCP Emergencies six (6) minutes into the event.

One minute after CCW is restored, the following conditions are noted by the crew:

* Containment sump level is rising slowly
* Loop II CCW surge tank is 5% and dropping rapidly

The proper course of action for the crew to take would be:

A. Commence a plant shutdown using 2102.004, Power Operations.

B. Start the Standby CCW Pump 2P33B and secure CCW Pump 2P33C.

C. Trip the Reactor and then secure the affected Reactor Coolant Pumps.

D. Override open the makeup valve to the expansion tank.

Answer:

C. Trip the Reactor and then secure the affected Reactor Coolant Pumps.

Notes:

A is incorrect because of the impending loss of all CCW to RCPs and the guidance found in 2203.025, Attachment A..
B is incorrect because this action would not affect system leakage.
D is incorrect because the makeup valve would already be full open.

References:

ANO-2-LP-RO-RCS, Objective 28
2203.025, Attachment A, Step 2 (RCP Emergencies)
2203.025, Attachment D, Step 1 (RCP Emergencies)

2203.012K, Annunciator 2K11 Corrective Actions (2K11-A1).
STM 2 - 43, Rev 5 Section 2.8 (Component Cooling Water System)

Historical Comments:

1/10/2002. Deleted "Remain at power" from distracter D due to feedback from the NRC that this could be a possible cue. BNC

QID:	0309	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.8 & 41.10	**10CFR55_43:**		**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	NATURAL CIRCULATION	**System Number**	A13	**K/A:**	AK1.2						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.2	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.5
Description:	Knowledge of the operational implications and the concepts contained in the normal, abnormal and emergency operating procedures associated with Natural Circulation Operations.										

Question:

Given the following:

* The Reactor has been tripped.
* All RCPs have been secured.
* LOCA Optimal Recovery Procedure has been entered.
* RVLMS level 3 indicates covered.
* RCS pressure equals 1800 PSIA.
* CET temperatures equal 610°F.
* RCS Cold Leg temperature, Tc, equals 582°F and lowering.
* RCS Hot Leg temperature, Th, equals 602°F and constant.

Which statement is correct concerning Natural Circulation?

A. Natural Circulation is not satisfied due to the CET/Th difference.

B. Natural Circulation is not satisfied due to the Th/Tc difference.

C. Natural Circulation is not satisfied due to the RVLMS level.

D. Natural Circulation is not satisfied due to Margin to Saturation.

Answer:

D. Natural Circulation is not satisfied due to Margin to Saturation.

Notes:

A is incorrect because CET/Th difference is less than 10 degrees F.
B is incorrect because the Th/Tc difference is less than 50 degrees F and both are constant or lowering.
C is incorrect because RVLMS level 4 is above the hot leg and will not impede natural circulation.

References:

Steam Tables
ANO-2-LP-RO-ELOCA, Objective 9
2202.003, Section 2, Step 11A (LOCA Recovery Procedure).

Historical Comments:

QID:	0310	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 41.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	EMERGENCY BORATION	**System Number**	024	**K/A:**	AK3.02						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	4.2	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.4
Description:	Knowledge of the reasons for the actions contained in the EOP that apply to Emergency Boration.										

Question:

Following a plant trip, Emergency Boration of the RCS is required.

When the CBOR attempts to close the Volume Control Tank (VCT) Outlet Valve (2CV-4873-1), the valve will not close.

At this point, at least one Boric Acid Makeup (BAM) Pump is verified to be aligned to the Charging Pump suction via the Emergency Borate valve (2CV-4916-2) to ensure a highly concentrated solution of boric acid solution is delivered to:

A. The RCS because the BAM pump discharge pressure is higher than the VCT pressure.

B. The RCS because the BAM pump discharge pressure is lower than the VCT pressure.

C. The VCT because the BAM pumps discharge through the mixing tee.

D. The VCT because the BAM pumps discharge directly into the VCT.

Answer:

A. The RCS because the BAM pump discharge pressure is higher than the VCT pressure.

Notes:

B is wrong because the BAM pump pressure is higher than VCT pressure.
C is wrong because the Emergency Borate Valve bypasses the mixing tee.
D is wrong because the BAM pump discharge does not go to the VCT when the Emergency Borate valve is open.

References:

ANO-2-LP-RO-ESPTA, Objective 18
2202.010, EOP Attachments - Exhibit 1
2202.032, Emergency Boration , Step 3
2203.032, Emergency Boration Technical Guide, Step 5

Historical Comments:

QID:	0311	**Rev:**	000	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00			
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	Modified	**Originator:**	Coble					
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE							
System Title:	Loss of Component Cooling Water	**System Number**	026	**K/A:**	AA1.05									
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.1	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.1			
Description:	Ability to operate and/or monitor the CCW Surge Tank, including level control and level alarms, and radiation monitors as they apply to the Loss of CCW													

Question:

Consider the following conditions.

* The plant is at 100% power.
* Loop II Component Cooling Water (CCW) Surge Tank level is slowly rising.
* The Loop II CCW Radiation Monitor alarm is in solid.
* Chemistry samples of Loop II CCW indicate short lived radionuclides.
* 2203.016, Excess RCS leakage AOP has been entered.

Given these conditions the Loop II CCW Surge Tank level should be maintained between __________ and Loop II CCW Surge Tank vent (2CV-5218) would be aligned to ____________.

A. 25% and 35%; atmosphere

B. 40% and 50%; atmosphere

C. 25% and 35%; the Gas Collection Header

D. 40% and 50%; the Gas Collection Header

Answer:

D. 40% and 50%; the Gas Collection Header

Notes:

The guidance found in the RCS Leakage AOP, Attachment A has the Surge Tank vent swapped to the GCH and level maintained between 40 and 50%. Thus D is the correct answer. The 25 - 35% range is within the makeup valve opening setpoints of 25 - 45%.

References:

ANO-2-LP-RO-EAOP, Objective 11
2203.016, Excess RCS Leakage - Attachment A
STM 2-43, Rev 5 (Component Cooling Water), 2.8.1

Historical Comments:

From in-house bank modification of QID 7007 from Unit 2 Operations Exam Bank.
1/17/2002- Spelled out GCH acronym based on NRC feedback.

QID: 0312 **Rev:** 000 **Rev Date:** 11/29/2001 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00

Lic Level: RS **Difficulty:** 4 **Taxonomy:** Ap **Source:** New **Originator:** Blanchard

10CFR55_41: 41.7 **10CFR55_43:** **Section:** 4.2 **Type:** Generic APE

System Title: Pressurizer Pressure Control System Malfunct **System Number** 027 **K/A:** AK2.03

RO Tier: 1 **RO Group:** 1 **RO Imp:** 2.6 **SRO Tier:** 1 **SRO Group:** 2 **SRO Imp:** 2.8

Description: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunction and its associated controllers and positioners.

Question:

Given the following plant conditions:

* Plant Power is 100%.
* All controllers are in their normal system lineup.
* Pressurizer Pressure is 2230 psia and controller selected to 'A'.
* Pressurizer Level is 65% and controller is selected to 'A'.
* No evolutions in progress.
* 2Y1 Power is lost and immediately restored.
* All components and controllers operate as designed.

With no operator action, which of the following is correct?

A. Both spray valves at 40% open, only backup heaters will be on.

B. Both spray valves closed, all backup and proportional heaters will be on.

C. Both spray valves closed, only proportional heater will be on.

D. Both spray Valves at 100% open, all backup and proportional heaters will be off.

Answer:

A. Both spray valves at 40% open, only backup heaters will be on.

Notes:

B and D are wrong because the backup heaters will automatically reset but the proportional heaters require operator action to reset.
C is wrong because the spray valve controller is programmed so that it will return the output of the controller to the pre-loss of power output which will cause the spray valves to go to a 40% open position based on RCS pressure.

References:

ANO-2-LP-RO-PZR, Revision 2, Objective 2
STM 2-3-1, Rev 5, Pressurizer Pressure & Level Control Systems, Section 2.2.2 and 2.2.5
2203.028, Rev 5, PZR System Malfunctions
2103.005, Rev 022-01-0, Pressurizer Operations, Section 6.0

Historical Comments:

QID:	0313	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Steam Line Rupture	**System Number**	040	**K/A:**	AK2.02						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	2.6	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	2.6
Description:	Knowledge of the interrelations between the Steam Line Rupture and its associated sensors and detectors										

Question:

Given the following:

* Reactor has been tripped for 10 minutes.
* SIAS, CCAS, CIAS, and MSIS have actuated.
* Containment temperature = 250 degrees F and rising.
* 'A' Steam Generator Pressure = 400 psia and lowering.
* 'B' Steam Generator Pressure = 900 psia and stable.
* Pressurizer Pressure = 1500 psia and slowly lowering.
* Pressurizer level = 18% and slowly lowering.
* All systems operate as designed.
* Annunciator 2K10-A6, CNTMT RAD HI, is actuated.
* Containment High Range Radiation monitors read 50 R/hr and rising.
* No other radiation monitor alarms are in or trending up.

Which of the following is the reason for the Containment High Radiation Alarm?

A. RCS leakage in addition to a Main Steam Line Break inside Containment.

B. RCS leakage inside Containment only.

C. Primary to Secondary leakage in addition to a Main Steam Line Break inside Containment

D. Main Steam Line Break inside Containment only.

Answer:

D. Main Steam Line Break inside Containment only.

Notes:

A and B are wrong because no low range radiation monitors are in alarm or trending up which would indicate RCS leakage.
C is wrong because a SGTR would not cause a large radiation monitor reading unless the core was uncovered and no low range monitors indicate activity.

References:

ANO-2-LP-RO-RMON Objectives 10 and 11
2203.012J, Annunciator 2K10 Corrective Actions for 2K10 -A6 (note above step 2.1)
STM 2-62, Radiation Monitoring System, Section 2.1.2

Historical Comments:

QID:	0314	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.8 & 41.10	**10CFR55_43:**		**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	RCS Overcooling	**System Number**	A11	**K/A:**	AK1.2						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.0	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.3

Description: Knowledge of the operational implications of normal, abnormal and emergency operating procedures as they apply to RCS Overcooling.

Question:

Given the following conditions:

* Reactor tripped, SIAS, CCAS, CIAS, CSAS and MSIS have actuated.
* RCS Pressure is 1475 psia and dropping.
* 'A' SG pressure is 900 psia and dropping slowly.
* 'B' SG pressure is 200 psia and dropping rapidly.
* All non-vital electrical busses are deenergized.
* All other systems are operating as designed.

Which of the following correctly describes how RCS temperature will be maintained post SG blowdown:

A. Depressurize the RCS and allow HPSI injection flow to cool the RCS

B. Bypass 'A' MSIV and open a SDBCS bypass valve to the condenser

C. Open the SG blowdown valves on 'A' SG and maintain SG level with 'A' EFW pump

D. Open the SDBCS Upstream ADV and throttle the Upstream ADV isolation on 'A' SG

Answer:

D. Open the SDBCS Upstream ADV and throttle the Upstream ADV isolation on 'A' SG

Notes:

A is incorrect due to HPSI filling the PZR and taking the RCS solid.
B is incorrect due to no circulating water available on LOOP.
C is incorrect due to the required large volume of water usage to feed and bleed a SG to remove decay heat with this method and ADV available.

References:

2202.005, Rev 005-01-0, Excess Steam Demand, Step 18.A
ANO-2-LP-RO-EESD, Rev 2, Excess Steam Demand, Objective 9.4.
2202.005, Rev 005-01-0, Excess Steam Demand Technical Guidelines, Step 18.0
STM 2-15, Rev 6, Steam Generators & Main Steam System, Section 3.2.3

Historical Comments:

QID:	0315	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	An	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.8	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Condenser Vacuum	**System Number**	051	**K/A:**	AK3.01						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	2.8	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.1
Description:	Knowledge of the reasons for the loss of steam dump capability upon loss of condenser vacuum.										

Question:

Given the following conditions:

* Reactor and turbine tripped.
* Loss of 2H1 and 2H2 busses.
* Condenser pressure is 6.2 in. HgA.
* Pressurizer pressure is 2100 psia.
* 'A' SG pressure 990 psia.
* 'B' SG pressure 990 psia.
* Steam Dump & Bypass Control System (SDBCS) master controller setpoint is 980 psia.
* All systems are operating as designed.
* No operator action has been taken.

Which of the following best describes the current RCS heat removal path:

A. Main steam safety valves.

B. Automatic SDBCS control with bypass valves to the main condenser.

C. Automatic SDBCS control with upstream atmospheric dump valves.

D. Automatic SDBCS control with downstream atmospheric dump valves.

Answer:

D. Automatic SDBCS control with downstream atmospheric dump valves

Notes:

A is incorrect due to steam pressure below the lowest MSSV of 1078psig.
B is incorrect due to condenser interlock.
C is incorrect due to valves being last in SDBCS program and are administratively controlled with permissive solenoid in off and HIC in manual with zero output.

References:

STM 2-23, Rev 6, Steam Dump & Bypass Control System, Sections 2.2 & 2.3.4 & 4.0
2203.012B, Rev 023-01-0, Annunciator 2K03 Corrective Actions (2K03- B14), Condenser interlock.
ANO-2-LP-RO-SDBCS, Rev 9, Steam Dump & Bypass Control System, Objective 11.
ANO-2-LP-RO-EAOP, Rev 5, Abnormal Operating Procedures, Objective 14.

Historical Comments:

QID:	0316	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Station Blackout	**System Number**	055	**K/A:**	EA1.06						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	4.1	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.5
Description:	Ability to operate and monitor the restoration of power with one EDG during a station blackout.										

Question:

Given the following:

* Reactor Tripped and SPTA's completed.
* All non-Vital and Vital AC busses deenergized.
* Station Blackout ORP has been entered.
* #1EDG has been successfully started.
* All associated equipment operates as designed.

Complete the following:
One Service water pump will / must be __________________ start(ed) and the associated EDG Service water outlet valve will /must be _________________ open(ed).

A. automatically; manually

B. manually; automatically

C. manually; manually

D. automatically; automatically

Answer:

D. automatically; automatically

Notes:

The Service Water Pump powered from the bus that the #1DG is started on will auto start and the EDG Service Water Outlet valve will auto open when power is restored.

References:

2202.008, Rev 005-01-0, Station Blackout, Step 11;
2202.008, Rev 005-01-0, Station Blackout Technical Guidelines, Step 11;
STM 2-42, Rev 13, Service Water & Aux Cooling Water Systems, Sections 3.1 and 3.5.8
ANO-2-LP-RO-ESBD, Station Blackout Lesson Plan, Objective 6.

Historical Comments:

QID:	0318	**Rev:**	000	**Rev Date:**	12/06/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	An	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Nuclear Service Water	**System Number**	062	**K/A:**	AA2.06						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.1

Description: Ability to determine and interpret the length of time after the loss of Service Water System flow to a component before the component may be damaged.

Question:

Given the following plant conditions:

* Reactor trip due to a Large Break LOCA.
* Lockout on 2A2 Bus.
* The Breaker for #2 EDG Service Water Outlet valve (2CV-1504-2) trips when opening
* #2 EDG scavenging air temperature is 200°F and going up.
* Annunciator 2K09-D1 "2DG2 Potential Engine Failure" is actuated.
* LOCA ORP entry section in progress.
* All other ES equipment operates as designed.

Which of the following actions must be taken to ensure no equipment damage with respect to #2 EDG:

A. Locally shutdown #2 EDG immediately.

B. Commence reducing loads on #2 EDG.

C. Bypass #2 EDG SW Outlet MOV (2CV-1504-2).

D. Start third SW pump and monitor #2 EDG operation.

Answer:

A. Locally shutdown #2 EDG immediately .

Notes:

"A" is correct because the LOCA ORP note states that EDG without SW could sustain damage within 3 minutes and that it scavenging air temperature rises above 170°F, the EDG must be secured.
"B" is incorrect since scavenging air temperature is already above the 170 F limit and direction to S/D the EDG is given in the procedure.
"C" is incorrect since there is no bypass on the EDG SW outlet valve.
"D" is incorrect since with the SW outlet valve closed, starting the third SW pump would be ineffective.

References:

2202.003, Rev 005-01-0, Loss Of Coolant Accident, Section 1, Step 9.B.3
2104.036, Rev 045-03-0, Emergency Diesel Generator Operations, Step 5.8.
ANO-2-LP-RO-EAOP, Abnormal Operating Procedures, Objective 17.

Historical Comments:

QID:	0319	**Rev:**	000	**Rev Date:**	12/06/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Plant Fire On-Site	**System Number**	067	**K/A:**	2.4.27						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.0	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.5
Description:	Knowledge of fire in the plant procedure.										

Question:

Given the following:

* Plant 100% steady state.
* Fire in 2A3 switchgear for 20 minutes.
* 2B5 damaged by explosion.
* Fire brigade responding.

In accordance with the AOP, which ONE (1) of the following activities is required to be taken based on these conditions?

A. Notify Health Physics (HP) for continuous coverage.

B. Manually trip the reactor and remain in Hot Standby.

C. Perform a plant shutdown and cooldown.

D. Reduce plant electrical loads to within S/U #3 capacity.

Answer:

C. Perform a plant shutdown and cooldown.

Notes:

'A' is not correct since 2A3 switchgear is not in controlled access.
'B' is not correct since the plant is stable and a controlled plant shutdown can be performed.
'C' is recommended per 2203.034, fire or explosion procedure and 2203.045, loss of 480 Volt Vital bus
'D' would have no consequence on the safe operation of the plant, in this event.

References:

AOP 2203.034, Rev 005, Fire Or Explosion, Steps 15 & 16
AOP 2203.045, Rev 000-00-0, Loss of 480 Volt Vital Bus, Step 10
ANO-2-LP-RO-EAOP, Abnormal Operating Procedures, Objective 25

Historical Comments:

QID: 0320 **Rev:** 000 **Rev Date:** 10/16/2001 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00

Lic Level: RS **Difficulty:** 3 **Taxonomy:** Ap **Source:** New **Originator:** Blanchard

10CFR55_41: 41.5 & 41.10 **10CFR55_43:** **Section:** 4.2 **Type:** Generic APE

System Title: Control Room Evacuation **System Number** 068 **K/A:** AK3.17

RO Tier: 1 **RO Group:** 1 **RO Imp:** 3.7 **SRO Tier:** 1 **SRO Group:** 1 **SRO Imp:** 4.0

Description: Knowledge of the reasons for the injection of boric acid into the RCS as they apply to the Control Room Evacuation.

Question:

Given the following plant conditions:

* Fire results in a control room evacuation.
* Immediate actions are completed successfully.
* RO #2 is performing follow-up actions and discovers the VCT outlet valve, 2CV-4873 has developed a hot short.

Which of the following actions should the RO #2 take:

A. The breaker to the VCT outlet valve should be opened and the valve opened.

B. The breaker to the VCT outlet valve should be opened and the valve closed.

C. The breaker to the VCT outlet valve should be closed and left in hot short position.

D. The VCT outlet valve should be isolated by a manual valve and breaker remains as is.

Answer:

B. The breaker to the VCT outlet valve should be opened and the valve closed.

Notes:

Note: a hot short causes the valve to fail in it’s undesirable position, in this case open.
'A' is wrong because with pressure in the VCT, proper boric acid will not be injected into RCS from BAMT gravity feed valves.
'B' is correct because the valve can not be close manually until it is deenergized.
'C' is incorrect because of same reason as 'A'; but it would be appropriate to open the valve breaker.
'D' is incorrect since there is no manually isolation for the VCT outlet valve.

References:

2203.014, Rev 014-06-0, Alternate Shutdown, RO2 Follow-up Actions, Step 12.
2203.014, Rev 014-06-0, Alternate Shutdown Technical Guidelines, Shorted Equipment Definition
P&ID M2231 Sheet 1, Rev 133
ANO-2-LP-RO-EAOP, Rev 05, Abnormal Operating Procedures, Objective 10.

Historical Comments:

QID: 0321 **Rev:** 000 **Rev Date:** 01/10/2002 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00

Lic Level: RS **Difficulty:** 3 **Taxonomy:** Ap **Source:** New **Originator:** Blanchard

10CFR55_41: 41.8 & 41.10 **10CFR55_43:** **Section:** 4.2 **Type:** Generic APE

System Title: Loss of Containment Integrity **System Number** 069 **K/A:** AK1.01

RO Tier: 1 **RO Group:** 1 **RO Imp:** 2.6 **SRO Tier:** 1 **SRO Group:** 1 **SRO Imp:** 3.1

Description: Knowledge of the operational implications of the effects of pressure on Containment leak rate as they apply to a loss of containment integrity.

Question:

Which of the following would contribute to the accumulated offsite dose the most during a Large Break LOCA, assuming all other systems/components operate as designed?

A. Failure of a single red train CAMs containment isolation valve to automatically close.

B. Failure to comply with the Tech Spec containment pressure/temperature requirements.

C. Failure of one containment air lock door to close following a surveillance.

D. Failure of one Spray Pump to start on a Containment Spray Actuation Signal (CSAS).

Answer:

B. Failure to comply with the Tech Spec pressure/temperature requirements.

Notes:

'A' is incorrect since the other valves are assumed to close;
'B' is correct since the TS P/T limits are assumed to be met so that design pressure is not exceeded during accident conditions
'C' is incorrect since TS requires only one air lock door to be closed to ensure the penetration is leak tight;
'D' is incorrect since only one spray pump/train is required, even though the second pump would help reduce containment pressure.

References:

Technical Specification 3.6.1.4, Amendment 225, Containment Pressure/Temperature Bases
Technical Specification 3.4.6.2.1, Amendment 226, Containment Spray System Bases
Technical Specification 3.6.1.3, Amendment 225, Containment Air Locks Bases
ANO-2-LP-WCO-CBLDG, Rev 10, Containment Building, Objectives 2, 16, 21

Historical Comments:

1/10/2002. Added "auto close" to distracter A and "containment pressure/temperature" to answer B based on NRC feedback. BNC

QID:	0322	**Rev:**	000	**Rev Date:**	09/11/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Inadequate Core Cooling	**System Number**	074	**K/A:**	EK2.03						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	4.0	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.0
Description:	Knowledge of the interrelations between the AFW pump and Inadequate Core Cooling.										

Question:

Given the following plant conditions:

* Reactor tripped.
* All RCP's secured.
* Loss of Main Feed Pump Lube Oil has occurred.
* 'A' & 'B' SG levels equal 5% narrow range.
* 'A' & 'B' SG pressures equal 910 psia.
* No feed is currently available to either SG.
* Loss of Feedwater EOP entered.
* All other systems operate as designed.

What is the preferred feed source:

A. Emergency Feedwater Pump 2P7A.

B. Emergency Feedwater Pump 2P7B.

C. Auxiliary Feedwater Pump 2P75.

D. Condensate Pump 2P2A

Answer:

B. Emergency Feedwater Pump 2P7B

Notes:

'A' is incorrect since the steam driven pump is not running and is assumed to have failed on overspeed and will take additional time to reset.
'B' is correct since it is motor driven and has throttle valves on it's discharge to each SG to minimize the possibility to damage feed ring on establishment of feeding SG's. This is the first pump that the EOP directs recovery of and is thus considered the preferred pump when it becomes available.
'C' is incorrect since it is non-vital powered.
'D' is incorrect since SG pressures are above shutoff head and a cooldown will be required before it can feed the SG's.

References:

2202.006, Rev 005-01-0, Loss of Feedwater, Step 11.D
2202.006, Rev 005-01-0, Loss of Feedwater Technical Guidelines, Step 11.D.
ANO-2-LP-RO-ELOSF, Rev 01, Loss of Feedwater, Objective 5

Historical Comments:

QID:	0323	**Rev:**	000	**Rev Date:**	12/04/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	High Reactor Coolant Activity	**System Number**	076	**K/A:**	AA2.02						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	2.8	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine/interpret corrective actions required for high fission product activity in RCS.										

Question:

Given the following plant conditions:

* Unit is operating at 100% power following a Plant Startup from a Refueling Outage.
* RCS Letdown Gross Rad Monitor (2RITS-4806A) reads 1E5 CPM and is slowly rising.
* Chemistry samples indicate that RCS activity is approaching the Technical Specification limits.

Which ONE of the following actions should be taken due to the rising RCS activity?

A. Bypass Letdown Demineralizers and swap the VCT inlet to the Hold Up Tanks.

B. Minimize Letdown to allow more dilution inventory from Charging into the RCS.

C. Increase Letdown flow to maximize RCS activity cleanup using demineralizers.

D. Isolate RCS Letdown Gross Rad Monitor (2RITS-4806A) to prevent over-ranging.

Answer:

C. Increase Letdown flow to maximize RCS activity cleanup using demineralizers.

Notes:

Tech Guide for High Activity in the RCS, AOP 2203.020 Step 3, states that raising Letdown flow will raise the rate of fission product removal and should offset any the raised dose seen by personnel due to the raised flow rate. This makes answer C correct. Answer A is wrong because it would only move the fission products to another tank which would have to be processed later. B is incorrect because too much inventory added to the RCS would violate Technical Specifications. D is incorrect because we want to continue to monitor RCS activity with this monitor.

References:

2203.020, Rev 007-05-0, High Activity in RCS, Step 3
2203.020, Rev 007-05-0, High Activity in RCS Tech Guidelines, Step 3
ANO-2-LP-RO-EAOP, Rev 05, Abnormal Operating Procedures, Objective 15.0

Historical Comments:

QID:	0324	**Rev:**	000	**Rev Date:**	10/16/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.8	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Dropped Control Rod	**System Number**	003	**K/A:**	AK1.10						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	2.6	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	2.9

Description: Knowledge of the operational implication of the definitions of Core Quadrant tilt as they apply to Dropped Control rod.

Question:

Given the following plant conditions:

* Plant is at 100% power.
* Control Element Assembly, (CEA) 047 drops to 100 inches withdrawn.
* CEA 047 is in regulating group 6 and is a target CEA for 'B' Core Protection Calculator (CPC).

COLSS calculated Azimuthal Tilt (AZ Tilt), will:

A. be a higher value than before the dropped CEA.

B. NOT change following the dropped CEA.

C. stop calculating following the dropped CEA.

D. be a lower value than before the dropped CEA.

Answer:

A. be a higher value than before the dropped CEA.

Notes:

'B' & 'D' are incorrect since the core tilt will be higher due to the reduced flux in the quadrant (fuel assembly) that has the dropped CEA.
'C' is incorrect since the incore calculated AZ tilt will only stop calculating below 15% power.
'A' is correct since the affected fuel assembly and core quadrant incores will detect the reduced flux and the increased flux in the rest of the core quadrant incores, therefore AZ tilt magnitude will be higher than initial calculated AZ tilt.

References:

COLSS STM 2-66, page22
Control Element Drive Mechanism Control System, STM 2-02, CEA locations
OP 2103.017, AZ power tilt Calc using the CPC system
ANO-2-LP-RO-COLSS, COLSS Operating System, Rev 10, Objective 19.2R

Historical Comments:

QID:	0325	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.8 & 41.10	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Reactor Trip	**System Number**	007	**K/A:**	EK1.06						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.7	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.1

Description: Knowledge of the operational implications of the relationship of emergency feedwater flow to S/G and decay heat removal following reactor trip.

Question:

Given the following:

* Reactor tripped two (2) hours ago.
* RCS temperatures are steady.
* RCS Tc is 545°F.
* SG pressures are steady at 1000 psia.
* SG levels are steady at 60% controlled by 'B' EFW pump and valves manually.
* Steaming is using upstream ADV's.
* Condenser vacuum is broken and MSIV's are closed.
* SG blowdown is secured.

Over the next two (2) hours, if SG levels are to remain constant, EFW flow rate:

A. will be raised.

B. will be reduced.

C. will stay the same.

D. will be reduced to zero.

Answer:

B. will be reduced.

Notes:

Decay heat value will lower by approximately one third over the next two hours; therefore EFW flow must be reduced by a corresponding amount.

References:

REF. 2202.010, standard attachments, Attachment 15, Condensate usage.
ANO-2-LP-RO-EFW, Emergency Feedwater System, Rev 05, Objective 15.

Historical Comments:

QID:	0326	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Pressurizer Vapor Space Accident	**System Number**	008	**K/A:**	AK2.02						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	2.7	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	2.7
Description:	Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the sensors and detectors.										

Question:

Given the following:

* Reactor tripped.
* Pressurizer Pressure equals 2400 psia.
* Quench tank temperature equals 180°F.
* Quench tank pressure equals 10 psig.
* All systems and components are operating as designed.
* Assume Pressurizer Relief Tailpipe temperature alarm is not present .

Which of the following PZR relief valve tail pipe temperatures indicate a code safety leak?

A. 240°F

B. 193°F

C. 180°F

D. 662°F

Answer:

A. 240°F

Notes:

Must convert quench tank pressure to absolute 10+15 = 25 psia
Must look up saturation pressure for 25 psia and interpolate between 20 and 30 psia.
250.34 - 227.96 = 22.38 ; 22.38/2 = 11.19
227.96 + 11.19 = 239.15
Therefore 240 °F is the most credible of the given choices.
Distracter 'D' is not a credible value given the above conditions since it is the saturation temperature for 2400 psia with 25 psia backpressure in the line. For that temperature to be seen, a partial blockage must occur in the tailpipe so that 2400 psia backpressure is seen on the leaking PZR relief valve.

References:

GEFES lesson plan AA50130-004; Obj 4.15
2203.016 step 8.C Rev. 009-01-0

Historical Comments:

QID:	0327	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Small Break LOCA	**System Number**	009	**K/A:**	EA1.16						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	4.2	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.2
Description:	Ability to operate and monitor Subcooling Margin Monitors as they apply to Small Break LOCA.										

Question:

Given the following plant conditions:

* Ten (10) minutes post reactor trip from 100% power.
* Containment radiation monitors read 1.2 REM/HR and rising.
* RCS pressure is 1850 psia and rising.
* Pressurizer level is 32% and rising.
* All RCP's are secured.
* Containment sump level is 50% and rising.
* Containment pressure is 14.6 psia and rising.

Which ONE (1) of the following should be performed for the given conditions?

A. Use RCS Tave and depressurize the RCS to maintain 30 - 45°F MTS.

B. Use RCS Th and depressurize the RCS to maintain 30 - 45°F MTS.

C. Use RCS Tc and depressurize the RCS to maintain 30 - 45°F MTS.

D. Use CET's and depressurize the RCS to maintain 30 - 45°F MTS.

Answer:

D. Use CET's and depressurize the RCS to maintain 30 - 45°F MTS.

Notes:

Standard attachments, Attachment 1, require the use of CET's to determine MTS when in natural circ and Th when RCP's are running. RCS Tc is used to monitor RCS cooldown rate only. Procedure 2203.016, floating step) requires depressurizing the RCS with a small break before starting a cooldown.

References:

2203.016, Rev 009-01-0, Excess RCS Leakage, Step 26 A (floating step)
2202.010, Rev 006-01-0, Standard Attachments, Attachment 1
ANO-2-LP-RO-EAOP, Abnormal Operating Procedures, Rev 05, Objective 11

Historical Comments:

QID:	0328	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Large Break LOCA	**System Number**	011	**K/A:**	EK2.02						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	2.6	**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	2.7
Description:	Knowledge of the interrelations between the pumps and Large Break LOCAs.										

Question:

Following a large break LOCA and a valid RAS, what is the status of the ECCS pumps, assuming all equipment operates as designed?

A. All HPSI pumps secured and all LPSI pumps running, with suction aligned to RWT.

B. Two HPSI pumps running and all LPSI pumps secured, with suction aligned to RWT.

C. All HPSI pumps secured and all LPSI pumps running, with suction aligned to Containment sump.

D. Two HPSI pumps running and all LPSI pumps secured, with suction aligned to Containment sump.

Answer:

D. Two HPSI pumps running and all LPSI pumps secured, with suction aligned to Containment sump.

Notes:

On a valid RAS, the LPSI pumps will automatically stop and the HPSI pumps will continue to run.

References:

STM 2-05, Emergency Core Cooling System, Section 3.1 Rev 09
ANO-2-LP-RO-ELOCA, LOCA EOP, Rev 05, Objective 3

Historical Comments:

QID:	0329	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Reactor Coolant Makeup	**System Number**	022	**K/A:**	AA1.08						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.4	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.3
Description:	Ability to operate and/or monitor VCT level as it applies to the Loss of Reactor Coolant Makeup.										

Question:

Given the following plant conditions:

* All components aligned in normal 100% power lineup.
* Reactor Makeup Water Flow Control Valve in "Dilute" position.
* A VCT level transmitter reference leg develops a leak.
* All systems / components operate as designed.

Which of the following actions should be completed:

A. Close Reactor Makeup Water Flow Control Valve, 2CV-4927, due to valve opening.

B. Isolate letdown flow due to loss of charging flow.

C. Switch VCT inlet/divert valve , 2CV-4826, to the VCT, due to valve opening to BMS.

D. Open VCT outlet valve, 2CV-4873-1 due to valve closing.

Answer:

C. Switch VCT inlet/divert valve , 2CV-4826, to the VCT, due to valve opening to BMS.

Notes:

VCT level transmitters will fail high on loss of reference leg.
'A' is incorrect due to Valve only opening if Reactor Makeup Water Flow Control Valve is in the 'Automatic' position.
'B' is incorrect since VCT level is not in the CCP stopping circuit, on actual low level in the VCT the low CCP suction pressure switch will stop the CCP, if SIAS is not present.
'C' is correct since the VCT inlet/divert valve will switch to the BMS position on a high level (78%)
'D' is incorrect since the RWT to CCP suction will not automatically open although actual VCT level will drop.

References:

2203.012L, Rev 030-01-0, Annunciator 2K12 Corrective Actions, 2K12 H-5 and G-5
2203.036, 006-03-0, Loss of Charging Technical Guidelines, Step 11
STM 2-04, Rev 12, Chemical and Volume Control system, Sections 2.3.7, 2.1.21.1,2.3.10
ANO-2-LP-RO-EAOP, Abnormal Operating Procedures, Rev 05, Objective 26

Historical Comments:

QID:	0330	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Residual Heat Removal System	**System Number**	025	**K/A:**	AA2.04						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.6
Description:	Ability to determine and interpret the location and isolability of leaks as they apply to the Loss of Residual Heat Removal System.										

Question:

Given the following:

* 'A' SDC Loop was placed in service 5 minutes ago with a normal system lineup.
* PZR level is 37% and dropping slowly.
* RCS temperature is 250°F and constant.
* PZR pressure is 260 psia and dropping slowly.
* Waste Tank (2T20) levels are constant.
* Containment sump level constant.
* Aux building sump and ESF room levels are not rising.
* Hold Up Tank (2T12) levels constant.
* SDC heat exchanger radiation monitor is not rising.

The most probable leak path is through the ______________________________________ and the correct action to take would be to ____.

A. 2P60A normal LPSI Recirc valve (2CV-5123-1); isolate 2CV-5123-1 flowpath to the RWT.

B. Letdown divert valve (2CV-4826); take 2CV-4826 handswitch to the VCT position.

C. 'A' SDC heat exchanger to Service Water; bypass and isolate the 'A' SDC heat exchanger.

D. 'A' ESF suction header relief (2PSV-5653); place 'B' SDC loop in service and secure 'A' loop.

Answer:

A. 2P60A normal LPSI Recirc valve (2CV-5123-1); isolate 2CV-5123-1 flowpath to the RWT.

Notes:

Distracter B is incorrect since 2T12 level is constant
Distracter C is incorrect since Radiation levels are constant out the SDC heat exchanger
Distracter D is incorrect since Aux building and ESF room drain level is constant and relief discharges to floor of 'A' ESF pump room.

References:

M2232 sheet 1, P&ID for Safety Injection system
M2236 sheet 1, P&ID for Containment Spray system
2104.004, Rev 028-02-0, Shutdown Cooling System, Step 7.17
ANO-2-LP-RO-EAOP, Abnormal Operating Procedure, Rev 5, Objective 22

Historical Comments:

1/10/2002. Added actions to be taken after diagnosis to this SRO only question based on feedback from the NRC. BNC

QID:	0331	**Rev:**	000	**Rev Date:**	10/16/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 41.10	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Anticipated Transient Without Scram	**System Number**	029	**K/A:**	EK3.11						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	4.2	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of the reasons for initialing emergency boration as they apply to ATWS.										

Question:

Given the following:

* A plant trip has occurred.
* SPTAs are in progress.
* CBOR notes that one CEA is not fully inserted.
* All Reactor Trip switchgear breakers are open.

The correct action to take based on these conditions would be to:

A. Continue to assess other safety functions because the Reactivity Safety Function is met.

B. Initiate emergency boration to ensure adequate shutdown margin is obtained.

C. Remove power to the CEA by depressing the DSS Emergency Trip pushbutton.

D. Remove Power to the CEA by de-energizing both MG sets for 10 seconds.

Answer:

B. Initiate emergency boration to ensure adequate shutdown margin is obtained.

Notes:

At ANO the SPTAs direct that emergency boration be implemented if all CEAs are not fully inserted. Emergency boration is implemented conservatively until there is a peer check on amount of CEAs inserted at which point emergency boration can be terminated. IAW the SPTAs step 3.B, the reactivity safety function can only be met if emergency boration is in progress. Power has already been removed from the CEA by the fact that the Reactor trip breakers are open.

References:

ANO-2-LP-RO-ESPTA, Revision 5, Objective 18
OP 2202.001, SPTAs, Revision 005-00-0, Step 3.B
EOP 2202.001, SPTA Technical Guide, Revision 5, Step 3

Historical Comments:

QID:	0332	**Rev:**	000	**Rev Date:**	10/16/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.0	**Type:**	Generic K/A				
System Title:	Steam Generator Tube Leak	**System Number**	037	**K/A:**	2.4.11						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.4	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.6
Description:	Knowledge of abnormal condition procedures.										

Question:

Given the following:

* RCS leakage to the "B" Steam Generator has been calculated to be 20 gpm.
* RCPs 2P32A and 2P32C are running.
* A plant shutdown has been completed IAW AOP 2203.038, Primary to Secondary Leakage.
* A plant cooldown is now directed by the AOP procedure.

The plant should be cooled down to below 520°F by _______ indication to ensure that ____ after the "B" Steam Generator is isolated.

A. Thot; Main Steam Safety valves do NOT lift.

B. Tcold; Main Steam Safety valves do NOT lift.

C. Thot; Margin to Saturation is maintained >30°F.

D. Tcold; Margin to Saturation is maintained >30°F.

Answer:

A. Thot; Main Steam Safety valves do NOT lift.

Notes:

As directed by steps 16 and 17 of AOP 2203.038, the RCS must be cooled down to less than 520°F by Thot indication because the SG pressure will go to Thot saturation pressure when it is isolated. The 520°F corresponds to a temperature below the lowest set pressure Main Steam Safety valves thus they will not lift. Margin to Saturation should always be maintained > 30°F if possible but this is based on RCS pressure at any RCS temperature.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 28
OP 2203.038, Primary to Secondary Leakage AOP, Revision 006-00-0, Steps 16 and 17
AOP 2203.038, Technical Guide, Revision 006-00-0, Step 16 and 17.

Historical Comments:

QID:	0333	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	An	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Steam Generator Tube Rupture	**System Number**	038	**K/A:**	EA2.08						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.4
Description:	Ability to determine viable alternatives for placing plant in a safe condition when condenser is not available when SGTR present.										

Question:

Given the following plant conditions:

* A complete Loss of Offsite Power occurs from 100% Power Operations.
* Both EDGs start and tie onto their respective safety buses.
* During SPTAs, RCS Pressure starts to drop rapidly and is currently 1575 psia.
* Steam Generator "A" level is 55% and rising rapidly.
* Steam Generator "B" level is 23% and being maintained with EFW.
* Main Steam Line "A" Radiation monitor readings are rising.

To place the plant in a safe condition and minimize any radioactive release will require the RCS to be COOLED down to less than 520°F by:

A. Backflowing the "A" Steam Generator to the RCS.

B. Steaming both Steam Generators to the atmosphere.

C. Steaming both Steam Generators to the Condenser.

D. Steaming the "B" Steam Generator to the atmosphere.

Answer:

B. Steaming both Steam Generators to the atmosphere

Notes:

Answer A is wrong because for the given conditions, backflowing the SGs to the RCS is not possible. This is the step to take after the cooldown to minimize the release of radioactive water. B is incorrect because the Condenser is not available due to the loss of circulating water and the inability to maintain a vacuum - condenser interlock prevents opening Steam Dump bypass valves to the condenser. D is incorrect because this will only cool down the B Steam Generator and the goal is to cool the ruptured Steam Generator below the saturation temperature for the lowest MSSV lift setpoint prior to isolation. Therefore answer C is the only viable option for cooling the plant down in preparation for Steam Generator isolation.

References:

ANO-2-LP-RO-ESGTR, Revision 3, Objective 5
OP 2202.004, SGTR EOP, Revision 005-00-0, Step 11
EOP 2202.004, Technical Guide, Step 11

Historical Comments:

QID:	0334	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 4.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Main Feedwater	**System Number**	054	**K/A:**	AK3.04						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	4.4	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of the reasons for the actions contained in the EOPs for Loss of MFW.										

Question:

Given the following plant conditions:

* Plant was operating at 100% Power.
* AFW Pump 2P75 is OOS for Pump bearing replacement.
* Both Main Feedwater Pumps tripped on low suction pressure.
* EFW Pump 2P7B fails to start.
* EFW Pump 2P7A trips on overspeed.
* CRS has completed SPTAs and diagnosed entry into 2202.006, Loss of Main Feedwater.
* All other components operate as designed.

Given these conditions, the crew should:

A. Secure all running RCPs to minimize the amount of heat input into the RCS.

B. Maintain two (2) RCPs running with one in each loop to balance SG heat removal.

C. Cooldown and depressurize the RCS to maximize margin to saturation.

D. Maximize steam generator blowdown to increase the heat removal rate from the RCS.

Answer:

A. Secure all running RCPs to minimize the amount of heat input into the RCS.

Notes:

With no feedwater available, the SG inventory would be rapidly depleted with the 4 MW of heat input to the RCS per RCP. The ORP for Loss of Feedwater directs securing of all RCPs thus preventing a more rapid loss of SG inventory due to this heat input.

References:

ANO-2-LP-RO-ELOSF, Revision 1, Objective 4
OP 2202.006, Loss of Feedwater ORP, Revision 005-01-0, Step 5.A
EOP 2202.006, Technical Guide, Revision 005-01-0, Step 5

Historical Comments:

1/10/2002. Changed distracter C to maximize verses minimize to make a more viable distracter based on NRC feedback. BNC

QID:	0336	**Rev:**	000	**Rev Date:**	10/16/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 41.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of DC Power	**System Number**	058	**K/A:**	AK3.01						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.4	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.7
Description:	Knowledge of the reasons for use of DC control power by D/Gs as they apply to the Loss of DC Power.										

Question:

Given the following plant conditions:

* Plant is operating at 100% Power.
* A complete loss of Red Train Vital DC Buses occur due to a fire involving 2D01.
* CRS directs crew to place 2DG1 in service on the 2A3 bus.

Which ONE of the following described the starting and stopping of 2DG1 during these conditions?

A. 2DG1 can be started from the control room but has to be secured locally.

B. 2DG1 has to be started locally but can be secured from the control room.

C. 2DG1 has to be started and stopped locally in the diesel room.

D. 2DG1 can be started and stopped remotely from the control room.

Answer:

C. 2DG1has to be started and stopped locally in the diesel room.

Notes:

To start and stop an EDG requires DC control power to energize the air start solenoids and the shut down relay to dump hydraulic fluid off the governor to place the fuel racks in the no fuel position. However, the air start solenoids can be manually overridden open locally and a backup source of 12 VDC power can be aligned for field flashing. This will get the EDG started. The manual pushbutton that actuates the mechanical overspeed shutdown device can be pushed locally to shut the engine down with a loss of DC. This explanation makes answers A, B and D wrong

References:

ANO-2-LP-RO-EDG, Revision 8, Objective 3
OP 2203.037, Loss of 125 VDC, Revision 003-04-0, Contingency Steps 33 A and C.
OP 2104.036, EDG Operations, Revision 045-03-0, Exhibit 1

Historical Comments:

QID:	0337	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Accidental Liquid Radwaste Release	**System Number**	059	**K/A:**	AA2.04						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.5
Description:	Ability to determine and interpret the valve lineup for a release of radioactive fluid as they apply to the Accidental Liquid Radwaste Release.										

Question:

Given the following:

* A Liquid Release Permit has been requested for Boric Acid Condensate Tank, 2T-69A.
* Chemistry has returned the permit to operations after sampling and analyzing the tank.
* While conducting the source check on the BMS Liquid Discharge Radiation Monitor, 2RE-2230 it is determined that the radiation monitor is not responding.
* The Shift Manager declares 2RE-2230 inoperable.

To prevent an accidental release of a non-permitted tank, the release of 2T-69A cannot continue unless:

A. the Plant Manager has approved the release with an inoperable radiation monitor.

B. independent verification of tank samples, release rate data, and lineup completed.

C. the inoperable radiation monitor, 2RE-2230, is returned to an operable status.

D. contingencies for analyzing grab samples every two (2) hours are established.

Answer:

B. independent verification of tank samples, release rate data, and lineup completed.

Notes:

In accordance with the requirements in the Offsite Dose Calculation Manual (ODCM) Specification L2.1.1, a liquid release of an onsite tank can continue with an inoperable radiation monitor as long as an independent sample is taken and analyzed to ensure release limits will not be exceeded. Also an inoperable radiation monitor requires an independent check of the proper valve lineup to ensure the sampled tank is the one released. Plant Manager approval is not required specifically for this case. His approval of plant procedures in general allows this exception. Grab samples during the release are not specified in the OCDM requirement nor the procedure.

References:

ANO-2-LP-SRO-TS, Revision 7, Objective 13
ODCM , Revision 13, Unit 2 Specification L2.1.1
OP 2104.014, LRW and BMS Operations, Revision 032-05-0, Supplement 1 - Step 2.9
OP 2104.014, LRW and BMS Operations, Revision 032-05-0, Supplement 3 - Step 12.0

Historical Comments:

1/10/2002, Reworded Stem to make question more like K&A statement. Deleted QID 363 due to its similarities to this question. These changes were based on NRC feedback. BNC

QID:	0338	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Accidental Gaseous Radwaste Release	**System Number**	060	**K/A:**	2.4.10						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.0	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.1
Description:	Knowledge of annunciator response procedures associated with an accidental gaseous radwaste release.										

Question:

Given the following plant conditions:

* Plant is in Mode 5 making preparations to refuel the reactor.
* RCS is in reduced inventory preparing to install SG nozzle dams.
* Containment Purge System is in service.
* When the 1st set of SG Manways are removed, the Control Room receives Annunciator 2K11 D-10 " Process Gas Radiation HI/LO".
* On 2C-25, the Gas Monitor for the Containment Purge System, 2RITS-8233, reading is above setpoint.
* Annunciator Corrective Action directs verification of Containment Purge secured.

The automatic actions that should have secured containment purge would be:

A. All three (3) Containment purge supply Isolations go closed.

B. Only the Outside-Outside purge supply and exhaust Isolations go closed.

C. Only the Inside-Inside purge supply and exhaust isolations go closed.

D. All three (3) Containment purge exhaust isolations go closed.

Answer:

B. Only the Outside-Outside purge supply and exhaust Isolations go closed.

Notes:

The only valves associated with the Containment Purge System that gets a closure signal on a high process radiation alarm is the Outside-Outside supply and exhaust valves. These valves are considered containment isolations and verified closed from the ESF control panels 2C-16 and 17. The closing of these valves will trip the exhaust fan on low suction pressure and the supply fan is interlocked to trip if the exhaust fan is not running.

References:

ANO-2-LP-RO-CVENT, Revision 8, Objective 13
OP 2203.012K, ACA for Process Gas Radiation High, Revision 029-04-0, Window 2K11 D-10
STM 2-9, Containment Cooling and Purge Systems, Revision 6, Sections 7.3, 7.5, 7.6 and Purge one line figure.

Historical Comments:

QID:	0339	**Rev:**	000	**Rev Date:**	10/10/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	ARM System Alarms	**System Number**	061	**K/A:**	AK2.01						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	

Description: Knowledge of the interrelationships between the Area Radiation Monitoring (ARM) System Alarms and the detectors at each ARM location.

Question:

Given the following plant conditions:

* The Plant is at 30% Power coming out of a refueling outage.

Which SET of the following radiation monitors, if in alarm, would provide the best indications of a primary to secondary leak? (Assume both are in alarm)

A. The N-16 monitor and condenser off-gas radiation monitor.

B. Main steam line radiation monitor and CCW loop II radiation monitor.

C. Auxiliary Building 335' elevation area monitor and the N-16 monitor

D. Loop I CCW radiation monitor and the condenser off-gas radiation monitor.

Answer:

A. The N-16 monitor and condenser off-gas radiation monitor.

Notes:

The N-16 monitors are much more sensitive at low power than the main steam line radiation monitors and along with the condenser offgas monitor, should provide positive indication of a SG tube leak. The CCW loop monitors and Aux Building area monition on elevation 335' should not be affected.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 28
OP 2203.038, Primary to Secondary Leakage AOP, Revision 006-00-0, Entry Conditions
STM 2-62, Radiation Monitoring System, Revision 6, Section 2.3

Historical Comments:

From in-house bank

QID:	0341	**Rev:**	000	**Rev Date:**	10/10/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Pzr Level Control Malfunction	**System Number**	028	**K/A:**	AA1.07						
RO Tier:	1	**RO Group:**	3	**RO Imp:**	3.3	**SRO Tier:**	1	**SRO Group:**	3	**SRO Imp:**	3.3

Description: Ability to operate and/or monitor charging pumps in maintenance of Pzr level as applied to the Pzr Level Control Malfunctions.

Question:

Given the following plant conditions:

* The plant is at full power.
* Pressurizer Level Control System master controller is in AUTO REMOTE.
* Pressurizer Level Control is selected to "CH 4627-A".
* Pressurizer Heater Low Level Cutout is selected to "A&B".
* Charging Pump Selector Switch, 2HS-4868, is in "A&B".
* Pressurizer Reference leg 2LT-4627-1 develops a leak.
* No operator action is taken.

WHICH ONE of the following describes the response of the Pressurizer Level Control System?

A. Charging Pumps A and B start, heaters energize, letdown flow decreases.

B. Charging Pumps A and B start, heaters cutout, letdown flow decreases.

C. Charging Pumps A and B get a stop signal, heaters energize, letdown flow increases.

D. Charging Pumps A, B, and C get a stop signal, heaters cutout, letdown flow increases.

Answer:

C. Charging Pumps A and B get a stop signal, heaters energize, letdown flow increases.

Notes:

The reference leg leak will cause a high indicated level input to the Pressurizer Level controller and associated bistables to cause level to indicate above setpoint by > 4.5%. This will in turn send a stop signal to the backup charging pumps in this case pumps A and B (the lead pump will continue to run), a signal to energize all pressurizer heaters and force the Letdown Flow Controller to maximum output.

References:

ANO-2-LP-RO-PZR, Revision 2, Objectives 9 and 10
STM 2-3-1, Pressurizer Pressure and Level Control, Revision 5, Sections 3.1 and 3.2
2103.005, Rev 21, Step 6.7 (Pressurizer Operations)

Historical Comments:

QID:	0342	**Rev:**	000	**Rev Date:**	10/10/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.8 & 41.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Offsite Power	**System Number**	056	**K/A:**	AK1.01						
RO Tier:	1	**RO Group:**	3	**RO Imp:**	3.7	**SRO Tier:**	1	**SRO Group:**	3	**SRO Imp:**	4.2

Description: Knowledge of the operational implications of the principle of cooling by natural convection as they apply to Loss of Offsite Power.

Question:

During a natural circulation cooldown, which ONE (1) of the following pressurizer level responses would indicate the presence of a void in the reactor vessel upper head?

A. A Pressurizer level increase when charging flow is directed through the auxiliary sprays.

B. A Pressurizer level decrease when charging flow is directed through the auxiliary sprays.

C. A Pressurizer level increase when charging flow is directed into the cold legs.

D. A Pressurizer level decrease when there is an increase in the cooldown rate.

Answer:

A. A Pressurizer level increase when charging flow is directed through the auxiliary sprays.

Notes:

Answer A is correct because a lowering of pressure in the pressurizer would cause expansion of the bubble in the head forcing water up into the pressurizer - just the opposite of answer B. Answer C is wrong because a level increase should be expected with charging going to the loops. Answer D is wrong because a cooldown should contract the RCS and lower Pressurizer level.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 9
OP 2203.013, Natural Circulation Operations, Revision 008-00-0, Step 25
AOP 2203.013, Technical Guide, Revision 008-00-0, Step 25

Historical Comments:

QID:	0343	**Rev:**	000	**Rev Date:**	11/06/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 4.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Instrument Air	**System Number**	065	**K/A:**	AK3.04						
RO Tier:	1	**RO Group:**	3	**RO Imp:**	3.0	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of the reasons for cross-over to backup air supplies as they apply to the Loss of Instrument Air.										

Question:

Given the following plant conditions:

* The plant is at 100% Power.
* Instrument Air (IA) pressure has started to degrade.
* All applicable steps of OP 2203.021, Loss of IA AOP have been taken up to this point.
* Unit 2 Instrument Air Header Pressure continues to drop and is currently 60 psig.

The correct course of action to take now would be:

A. Close the cross-connect with Unit 1 to prevent a loss of IA on Unit 1.

B. Open the cross-connect with Unit 1 to prevent a loss of IA on Unit 2.

C. Contact Unit 1 to start additional IA compressors to prevent a loss of IA on Unit 1.

D. Cross-connect the Service Air System to IA to prevent a loss of IA on Unit 2.

Answer:

A. Close the cross-connect with Unit 1 to prevent a loss of IA on Unit 1.

Notes:

The guidance found in the Loss of IA AOP directs the crew to close the cross connect with Unit 1 at less than 60 psig to prevent loosing IA to both units.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 16
STM 2-48, Instrument Air, Revision 3, Section 3.2
OP 2203.021, Loss of IA AOP, Revision 008-01-0, Entry Conditions and step 3.

Historical Comments:

QID:	0344	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.1	**Type:**	Reactivity Control				
System Title:	Control Rod Drive System	**System Number**	001	**K/A:**	A4.03						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	4.0	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to manually operate and/or monitor in the control room CRDS mode control.										

Question:

Given the following plant conditions:

* A Shutdown Bank Control Element Assemblies (CEAs) have been pulled to their "Full Out Position".
* The CBOR is pulling Regulating Group CEAs using the "Manual Sequential", MS, mode during a Reactor Startup.
* CEA Regulating group 3 is currently at 108.7 inches withdrawn.
* No other CEA mode of operation has been used by the CBOR during this Reactor Startup.
* No CEAs have been repositioned to minimize finger wear in accordance with 2102.004 Attachment D, Programmed CEA Insertion to Minimize CEA Finger Wear.
* All systems operate as designed.

At this point CEA Regulating Group 2 will be at the _________________________ and CEA Regulating Group 4 should be at __________ inches withdrawn.

A. Upper Electrical Limit; 10

B. Upper Group Stop; 10

C. Upper Electrical Limit; 60

D. Upper Group Stop; 60

Answer:

B. Upper Group Stop; 10

Notes:

In the Manual Sequential Mode of operations, the Upper Group Stop (145 Inches withdrawn) will prevent further CEA movement to the Upper Electrical Limit (150 inches withdrawn). This makes answers A and C wrong.

The next subsequent groups begins to move out when the previous group reaches the Upper Sequential Permissive at 98.7 inches withdrawn so group 4 would be at 10 inches withdrawn at 108.7 inches withdrawn on group 3. This makes D wrong.

References:

ANO-2-LP-RO-CEDM Objective 7
STM 2-02, Rev 7, Control Element Drive Mechanism Control System, Sections 1.2 and 4.2.1.4
OP 2102.016, Reactor Startup, Revision 007-02-0, Step 8.3C
OP 2103.004, Power Operations, Revision 027-05-0, Step 8.21.3 and Attachment D

Historical Comments:

QID:	0345	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.1	**Type:**	Reactivity Control				
System Title:	Control Rod Drive System	**System Number**	001	**K/A:**	K4.20						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.2	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.4
Description:	Knowledge of CRDS design feature(s) and/or interlocks which provide the permissives and interlocks associated with increase from zero power.										

Question:

Given the following plant conditions:

* The CBOR is pulling Control Element Assemblies (CEAs) during a reactor startup.
* Reactor Power is 1E-2%.

Which ONE of the following describes the CEDMCS CEA Withdrawal Prohibit (CWP) function during the Reactor Startup?

A. Initiated by CEA Pulse Counter System of the Plant Computer due to OUT OF SEQUENCE condition and prohibits ALL OUTWARD movement of ALL CEAs.

B. Initiated by Core Protection Calculators (CPCs) due to OUT OF SEQUENCE condition and prohibits ALL movement of ALL CEAs.

C. Initiated by Core Protection Calculators (CPCs) due to PRETRIP conditions on Low DNBR and prohibits ALL OUTWARD group movement of Regulating Groups 1-6 CEAs.

D. Initiated by CEA Pulse Counter System of the Plant Computer due to PRETRIP conditions on Low DNBR and prohibits ALL group movement of Regulating Groups 1-6 CEAs.

Answer:

C. Initiated by Core Protection Calculators (CPCs) due to PRETRIP conditions on Low DNBR and prohibits ALL OUTWARD group movement of Regulating Groups 1-6 CEAs.

Notes:

A is wrong because outward motion of individual CEAs can still be performed in Manual Individual Mode of operations. B is wrong because inward movement can still be performed on all CEA Groups. D is wrong because inward movement can still be performed on CEA Groups 1-6. Answers A and D are also wrong because the signal for a CWP does not come from the CEA pulse counters.

References:

ANO-2-LP-RO-CEDM, Rev 08, Objectives 3 and 7
STM 2-02, Rev 07, Control Element Drive Mechanism Control System, Sect 4.2.1.1, 4.2.1.2, and 4.2.1.3
STM 2-65-1, Rev 06, Core Protection Calculator System, Section 2.5
2203.012J, Rev 028-04-0, Annunciator 2K10 Corrective Action, Window 2K10-B1

Historical Comments:

QID:	0346	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Reactor Coolant Pump System	**System Number**	003	**K/A:**	K6.04						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.8	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.1
Description:	Knowledge of the effect of a loss or malfunction of containment isolation valves will have on RCP operation.										

Question:

Given the following plant conditions:

* The plant is at 100% power.
* An inadvertent CIAS occurs.
* Crew has entered 2203.039, Inadvertent CIAS.
* CCW to Reactor Coolant Pumps (RCPs) CANNOT be restored within 10 minutes of event initiation.
* Crew trips the plant and secures all RCPs.

The correct action to take next to prevent RCP seal failures due to heat buildup would be to:

A. Start all available Containment Cooling Units aligned to Service Water.

B. Commence a rapid plant cooldown to Shutdown Cooling entry conditions.

C. Verify RCP Controlled Bleedoff is isolated to the VCT and Quench Tank.

D. Make a Containment entry and Isolate RCP vapor seal leakoff to the RDT.

Answer:

C. Verify RCP Controlled Bleedoff is isolated to the VCT and Quench Tank.

Notes:

The answer to this question is based on the caution and step 4E in 2203.039 which requires the controlled bleedoff path to be secured to prevent RCS water flowing up through the RCP seal package thus eliminating the heat source to the seals. A and C are options that may be taken later to cool the RCS/Containment but this will not prevent RCP seal failures from occurring in the near term. D would only isolate a very small flowpath back to the RDT but the major flowpath would still be available.

References:

ANO-2-LP-RO-EAOP, Objective 29
STM 2-3-2, Rev 5, RCPs, Section 1.5
2203.039, Rev 003-02-0, Inadvertent CIAS, Step 4
2203.039, Rev 003-02-0, Inadvertent CIAS Technical Guide, Step 4

Historical Comments:

QID:	0347	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.2 to 41.9	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Reactor Coolant Pump System	**System Number**	003	**K/A:**	K1.13						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of the physical connections and/or cause-effect relationships between the RCPs and RCP bearing lift oil pumps.										

Question:

Which of the following is an interlock associated with the Reactor Coolant Pump (RCP) and RCP Oil Lift Pump System?

A. The RCP will not start unless RCP Oil Lift Pump has been running for three minutes.

B. The RCP Oil Lift Pump will stop one minute after the RCP has started.

C. The RCP will not start unless RCP Lift Pump pressure is greater than 400 psig.

D. The RCP Oil Lift Pump will stop when the RCP hand switch is placed in STOP.

Answer:

C. The RCP will not start unless RCP Lift Pump pressure is greater than 400 psig.

Notes:

A and C are procedural requirements of the RCP startup procedure 2103.006 and not automatic interlocks.
D is wrong because the lift pump will start when the handswitch is placed in STOP.

References:

ANO-2-LP-RO-RCS, Objective 8
STM 2-3-2, Rev 5, RCPs, Sections 1.7 and 1.7.1
2103.006, Rev 014-00-0, RCP Operations, Steps 6.1, 7.8, and 7.9

Historical Comments:

QID:	0348	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.1	**Type:**	Reactivity Control				
System Title:	Chemical and Volume Control System	**System Number**	004	**K/A:**	A3.06						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.9	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.8
Description:	Ability to monitor automatic operation of the CVCS including effects of Tave and Tref.										

Question:

The plant status is as follows:

* Reactor power is 80% and stable.
* Tavg = Tref at 570°F.
* Pressurizer Level indicates 50%.
* Pressurizer Level Controller is in Remote Auto.
* Letdown Flow Controller is in Automatic.
* All three Charging Pump hand switches are in Automatic.
* Charging Pump Selector switch is in the "B & C" position.

The correct alignment regarding the status of Charging and Letdown for this condition would be: (Reference Material Provided)

A. Three (3) Charging Pumps will be running with maximum letdown flow

B. Two (2) Charging Pumps will be running with maximum letdown flow

C. Three (3) Charging Pumps will be running with minimum letdown flow

D. Two (2) Charging Pumps will be running with minimum letdown flow

Answer:

C. Three (3) Charging Pumps will be running with minimum letdown flow

Notes:

B and D are wrong because the level deviation between the actual pressurizer level and the pressurizer level setpoint for the above stated conditions is ~ 6%. This would start both backup charging pumps along with the running lead Pump A.
A would be wrong because the level deviation would also make the letdown flow controller output go to its minimum automatic control level output to attempt to raise Pressurizer level.

This question will requires 2102.004 Attachments C and E to be used as a reference.

References:

ANO-2-LP-RO-PZR, Objectives 9 and 10
STM 2-3-1,Rev 5, Pzr Pressure and Level Control, Sections 3.2.4, 3.2.5, 3.2.6, 3.2.7, 3.2.8, and 3.2.9
2103.005, Rev 022-01-0, Pressurizer Operations, Steps 6.8 and 6.9

2102.004, Rev 027-04-0, Power Operations, Attachments C and E

Historical Comments:

QID:	0349	**Rev:**	000	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.1	**Type:**	Reactivity Control				
System Title:	Chemical and Volume Control System	**System Number**	004	**K/A:**	K4.13						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.1	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	

Description: Knowledge of CVCS design feature(s) and/or interlocks which provide for interlock between letdown isolation valve and flow control valve.

Question:

Given the following plant conditions:

* Plant is operating at 100% Power.
* Charging Pump 2P-36C is running.
* Charging Pump Selector Switch is in "A & B" position.
* Charging Pump 2P-36C trips.

With no operator action, the first automatic action that will occur to protect the CVCS Regen Heat Exchanger from excessive thermal stresses would be:

A. Charging Pump 2P-36A will start immediately to supply cooling flow to the Regen Heat Exchanger.

B. Charging Pump 2P-36B will start immediately to supply cooling flow to the Regen Heat Exchanger.

C. Letdown System Isolation valve (2CV-4820-2) will CLOSE on Regen Heat Exchanger Outlet temperature high.

D. Regen Heat Exchanger Inlet Isolation valve (2CV4821-1) will CLOSE on Regen Heat Exchanger Letdown outlet temperature high.

Answer:

C. Letdown System Isolation valve (2CV-4820-2) will CLOSE on Regen Heat Exchanger Outlet temperature high.

Notes:

A and B are wrong because the neither backup Charging Pump will auto start until Pzr level deviates from setpoint by 3.1% and 4.8% respectively.
D is wrong because the Heat Exchanger Inlet valve (2CV-4821-1) only closes on a CIAS and SIAS.

References:

ANO-2-LP-RO-CVCS, Rev 8, Objective 4
STM 2-04, Rev 12, Chemical and Volume Control System, Sections 2.1.2, 2.1.3, 2.1.4, 2.1.6 and 2.1.10
2104.002, Rev 038-06-0, Chemical and Volume Control, Section 6.0
2203.012L, Rev 030-01-0, Annunciator 2J12 Corrective Actions, 2K12-B1.

Historical Comments:

QID:	0350	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00

Lic Level: RS **Difficulty:** 3 **Taxonomy:** C **Source:** New **Originator:** Coble

10CFR55_41: 41.7 **10CFR55_43:** **Section:** 3.2 **Type:** RCS Inventory Control

System Title: Engineered Safety Features Actuation System **System Number** 013 **K/A:** A4.01

RO Tier: 2 **RO Group:** 1 **RO Imp:** 4.5 **SRO Tier:** 2 **SRO Group:** 1 **SRO Imp:** 4.8

Description: Ability to manually operate and/or monitor in the control room ESFAS initiated equipment which fails to actuate.

Question:

Given the following plant conditions:

* A plant trip has been initiated due to a Steam Generator Tube Rupture.
* RCS pressure is 1590 psia and slowly dropping.
* Containment pressure is 14.7 psia and stable.
* All systems are operating as designed.

Which ONE of the following ESF actions should have occurred?

A. Letdown Header Isolation valve 2CV-4820-2 should have CLOSED.

B. Letdown Regen HX Outlet valve 2CV-4823-2 should have CLOSED.

C. RCP CCW Supply Isolation valve 2CV-5236-1 should have CLOSED.

D. RCP CCW Return Isolation valve 2CV-5255-1 should have CLOSED.

Answer:

A. Letdown Header Isolation valve 2CV-4820-2 would be CLOSED.

Notes:

For the given condition, only an SIAS should be initiated. B, C, and D receive a CIAS and not a SIAS making them wrong.

References:

ANO-2-LP-RO-ESFAS, Rev 07, Objective 3
STM 2-70, Rev 07, ESFAS Actuation Description Table
STM 2-04, Rev 12, CVCS, Figure 1, Sections 2.1.2 and 2.1.5
STM 2-43, Rev 5, Section 3.2.18 and one line figure of CCW.
2202.010, Rev 006-01-0, Standard Attachments, Attachment 2 and Attachment 5

Historical Comments:

QID:	0351	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	An	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.2	**Type:**	RCS Inventory Control				
System Title:	Engineered Safety Features Actuation System	**System Number**	013	**K/A:**	K3.03						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	4.3	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	4.7
Description:	Knowledge of the effect that a loss or malfunction of the ESFAS will have on the Containment.										

Question:

Given the following plant conditions:

* Containment Spray Pump 2P35A is tagged out for motor replacement.
* A large break LOCA has occurred.
* Containment Spray Pump 2P35B starts and is providing 2000 gpm flow to Containment.
* RCS pressure is 150 psia and dropping.
* Crew has entered OP 2202.003, Loss of Coolant Accident.
* Loop 2 Service Water Supply Header Isolation to Containment Coolers, 2CV-1510-2, has failed to OPEN.

The amount of safety related equipment required to maintain adequate Containment temperature and pressure control :

A. is available due to adequate flow from the Green Train of Containment Spray and the Green Train of Containment Cooling units.

B. is NOT available due to inadequate flow from the Green Train of Containment Spray and the Green Train of Containment Cooling units.

C. is available due to adequate flow from the Green Train of Containment Spray and the Red Train of Containment Cooling units.

D. is NOT available due to inadequate flow from the Green Train of Containment Spray and the Red Train of Containment Cooling units.

Answer:

C. is available due to adequate flow from the Green Train of Containment Spray and the Red Train of Containment Cooling units.

Notes:

The Containment Spray system (CSS) in conjunction with the Containment cooling system (CCS) provides sufficient redundancy so that any of the following combinations of equipment will provide adequate heat removal to attenuate the post accident pressure and temperature conditions imposed upon the Containment following a LOCA or Main Steam Line Break (MSLB):

* all four Containment cooling units; or
* both loops of the CSS; or
* two of the four Containment cooling units and one CSS loop

This makes answers B and D wrong. Answer A is wrong because Loop 2 SW to Containment serves the Green Train Containment Cooling Units and is not available.

References:

STM 2-08, Rev 07, Containment Spray System, Section 1.3
STM 2-09, Rev 06, Containment Cooling and Purge Systems, Section 2.7
STM 2-42, Rev 13, Service Water & Aux Cooling Water Systems, Section 3.5.4
2203.003, Rev 005-01-0, LOCA Recovery Procedure, Step 8 of the Safety Function Status Check
2203.003, Rev 005-01-0, LOCA Recovery Technical Guide

Historical Comments:

QID:	0352	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.1	**10CFR55_43:**		**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Nuclear Instrumentation	**System Number**	015	**K/A:**	2.2.12						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.0	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.4
Description:	Knowledge of surveillance procedures associated with Nuclear Instrumentation										

Question:

Given the following:

* Plant power is 85% and power ascension in progress.
* You are performing the Power Distribution channel checks in accordance with the CBO Power Distribution and Burn up Log to ensure compliance with Technical Specifications.

For the following to be considered operable Excore Linear Power, CPC Neutron Power, and CPC Delta T Power have to be within ________ of ____________________.

A. ± 2%; Turbine First Stage Power

B. ± 2%; COLSS Calorimetric Power

C. -0.5/+10%; Turbine First Stage Power

D. -0.5/+10%; COLSS Calorimetric Power

Answer:

B. ± 2%; COLSS Calorimetric Power

Notes:

A is incorrect because Turbine First Stage Power is not the power used for comparison.
B is correct because Note 2b of T.S. Table 4.3-1 requires the three powers listed above to be within the absolute value of 2% of the COLSS Calorimetric Power when above 80% power.
C & D are valid distracters because the limits used are valid for plant power between 15% and 80% but incorrect because of the stem conditions.

References:

ANO-2-LP-RO-NI, Rev 09, Nuclear Instrumentation System, Objective 8
ANO-2 Technical Specifications, Amendment 186, Table 4.3-1 - Note 2b
1015.003B, Rev 046-00-0, Unit Two Operations Logs, Attachment B Page 9
STM 2-67-1, Rev 03, Excore Nuclear Instrumentation, Section 3.1

Historical Comments:

QID: 0353 **Rev:** 000 **Rev Date:** 12/08/2001 **RO Select:** Yes **SRO Select:** No **Points:** 1.00

Lic Level: R **Difficulty:** 2 **Taxonomy:** K **Source:** New **Originator:** Coble

10CFR55_41: 41.7 **10CFR55_43:** **Section:** 3.7 **Type:** Instrumentation

System Title: Nuclear Instrumentation System **System Number** 015 **K/A:** K2.01

RO Tier: 2 **RO Group:** 1 **RO Imp:** 3.3 **SRO Tier:** **SRO Group:** **SRO Imp:**

Description: Knowledge of the bus power supplies to the nuclear instrumentation channels, components and interconnections.

Question:

Given the following plant conditions:

* Plant is in Mode 3 coming out of a refueling outage.
* 120 VAC Electrical Bus 2RS2 is inadvertently de-energized.

Which ONE of the following instruments would be de-energized?

A. Startup Channel Number 1 indications.

B. Startup Channel Number 2 indications.

C. Channel A Excore Log Safety Channel indications.

D. Channel D Excore Log Safety Channel indications.

Answer:

B. Startup Channel Number 2 indications

Notes:

Channel A Log Safety Power indications and Startup Channel Number 1 indications are powered from 2RS1.
Channel D Log Safety Channel indications are powered from 2RS4.

References:

ANO-2-LP-RO-NI, Rev 9, Excore Nuclear Instrumentation, Objective 4
STM 2-67-1, Rev 3, Excore Nuclear Instrumentation, Section 2.3.1
2203.012J, Rev 028-03-0, Annunciator 2K10 Corrective Actions for 2K10-K5.
Electrical Print E-2456, Rev 28, Sheet 4
Electrical Print E-2727, Rev 6, Sheet 8

Historical Comments:

QID:	0354	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.7	**Type:**	Instrumentation				
System Title:	In Core Temperature Monitoring System	**System Number**	017	**K/A:**	A1.01						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.7	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.9

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM System controls including Core Exit Temperatures.

Question:

Given the following plant conditions:

* Both Reactor Vessel Level Monitoring System (RVLMS) Channels have been declared inoperable due to number of sensors available.
* A Small Break Loss of Coolant Accident occurs along with a Loss of Offsite Power.
* Emergency Diesel Generator 2DG1 has tripped on Generator Differential.
* All other systems operated as designed.

Which ONE of the following correctly states how Core Exit Thermocouples (CETs) are used to determine if the Core Heat Removal safety function is adequately satisfied?

A. All CET temperatures indicate less than 1000 °F and not rising.

B. Average of all CETs indicate less than 50 °F superheated.

C. All CET temperatures indicate less than 700 °F and not rising.

D. Average of all CETs indicate less than 25 °F superheated.

Answer:

C. All CET temperatures indicate less than 700 °F and not rising.

Notes:

CET superheat needs to be less than 10 °F to satisfy the Core Heat Removal Safety Function. 700 °F constant or lowering was the setpoint chosen by the CE owners group as the limit to ensure core uncovery was not progressing.

References:

2202.003, Rev 005-01-0, LOCA Recovery Procedure, Safety Function Status Check - Step 5
2202.009, Rev 005-01-0, Functional Recovery Procedure, HR-2, Step 63
2202.009, Rev 005-01-0, Functional Recovery Technical Guide, HR-2, Step 63
ANO-2-LP-RO-RVLMS, Rev 6, RVLMS, Objective 8
STM 2-75, Rev 0, Reactor Vessel Level Monitoring System, Sections 2.1.1, 2.2, 2.2.1, 2.5
STM 2-76, Rev 2, Safety Parameter Display System (SPDS), Page 12

Historical Comments:

QID:	0355	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.7	**Type:**	Instrumentation				
System Title:	Incore Temperature Monitoring System	**System Number**	017	**K/A:**	K6.01						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.7	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.0
Description:	Knowledge of the effect of a loss or malfunction of the ITM System sensors and detectors.										

Question:

Given the following plant conditions:

* The plant is at 100% power.
* The Monthly Post Accident Channel Checks surveillance is being performed on the Reactor Vessel Level Monitoring (RVLM) instrument on 2C388-1 in accordance with OPS-B18 Log.

For the RVLM instrument to be considered operable, a minimum of _____ level sensors must be operable in the upper plenum of the Reactor vessel and a minimum of _____ level sensors must be operable in the dome region of the Reactor vessel.

A. 1; 2

B. 2; 1

C. 2; 2

D. 1; 3

Answer:

B. 2; 1

Notes:

The TS bases for T.S. 3.3.3.6, Post Accident Monitoring requires 2 operable sensor in the upper plenum region of the core and 1 operable sensors in the dome region of the core.

References:

ANO-2-LP-RO-RVLMS, Rev 6, Reactor Vessel Monitoring System, Objective 11
STM 2-75, Rev 0, Reactor Vessel Monitoring System, Section 1.2
2105.003, Rev 006-02-0, RVLMS Operations, Section 3.0 and Attachment A
T.S. 3.3.3.6 Bases, Amendment No. 191
OPS-B18 Log, Rev 6/11/01

Historical Comments:

QID:	0356	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	An	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.2 to 41.9	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Containment Cooling System	**System Number**	022	**K/A:**	K1.01						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.5	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.7

Description: Knowledge of the physical connections and/or cause-effect relationships between the Containment Cooling System and Service Water System/Cooling System.

Question:

Given the following plant conditions:

* Plant has tripped from 100% power due to a Loss of Offsite Power.
* Two Main Steam Safety Valves stick open on 'B' Steam Generator (SG).
* RCS pressure is currently 1800 psia and dropping.
* 'B' SG Pressure is 600 psia and dropping rapidly.
* 'A' SG Pressure is 800 psia and dropping slowly.

Which ONE of the following correctly describes the response of the Service Water Isolations to/from the Containment Cooling Units after power is restored?

A. The supply isolations open immediately and the return isolations will open after the supply isolations start opening.

B. The supply isolations and return isolations open immediately together.

C. The supply isolations open after a time delay and the return isolations open after the supply isolations start opening.

D. Service Water Isolations do not open because they only actuate open on a Containment Cooling Actuation Signal.

Answer:

C. The supply isolations open after a time delay and the return isolations open after the supply isolations start opening.

Notes:

A is wrong because there is a time delay associated with the inlet valve after a LOOP.
B is wrong because the outlet valve will not start opening until the inlet valve is 10% open.
D is wrong because a the valves also come open on an MSIS actuation.

References:

ANO-2-LP-RO-CVENT, Rev 08, Containment Ventilation System, Objective 17
ANO-2-LP-RO-SWACW, Rev 10, SW & ACW Systems, Objective 19
STM 2-9, Rev 6, Containment Cooling and Purge Systems, Sections 2.7, 2.7.1, and 2.7.2
2104.029, Rev 051-05-0, Service Water System Operations, Section 3.0

Historical Comments:

8/15/01 Developed from Ops Exam Bank Question (QID 8813) with major modifications.

QID:	0357	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Containment Cooling System	**System Number**	022	**K/A:**	K3.02						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.0	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.3
Description:	Knowledge of the effect that a loss or malfunction of the Containment Cooling System will have on Containment Instrumentation readings.										

Question:

Given the following plant conditions:

* Plant is at 100% Power in the middle of July.
* Main Chill Water Return from Containment 2CV-3851-1 fails closed.
* Containment Pressure rises from 13.97 psia to 14.35 psia.
* Containment Temperature rises from 117.5 °F to 122.3 °F.

Instrumentation inside Containment that have recently been identified to demonstrate erratic behavior based on the temperature and pressure changes above are:

A. Steam Generator pressure instruments

B. Pressurizer pressure instruments

C. Low Range radiation monitors

D. Reactor Coolant System temperature instruments

Answer:

D. Reactor Coolant System temperature instruments

Notes:

Erratic behavior of RCS temperature instrument were noticed during a 14 day Containment Fan Cooler Operability test during July of 2001 due to warmer Service Water heating up the Containment. This caused an erratic trend in a concurrent RCS leakrate procedure. Pressure instruments should not be affected due to a vacuum reference leg. High Range radiation monitors have been identified to be inaccurate in a harsh Containment environment but not the Low Range monitors.

References:

ANO-2-LP-AO-CVENT, Rev 08, Containment Ventilation Systems, Objective 19
2104.033, Rev 040-06-0, Containment Atmosphere Control, Supplement 3 Note above Step 1.0
CR-ANO-2-2001-0607
Unit 2 OPS Lessons Learned

Historical Comments:

QID:	0358	**Rev:**	001	**Rev Date:**	12/04/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00

Lic Level: RS **Difficulty:** 3 **Taxonomy:** C **Source:** Modified **Originator:** Coble

10CFR55_41: 41.5 **10CFR55_43:** 43.5 **Section:** 3.4 **Type:** RCS Heat Removal

System Title: Condensate System **System Number** 056 **K/A:** A2.04

RO Tier: 2 **RO Group:** 1 **RO Imp:** 2.6 **SRO Tier:** 2 **SRO Group:** 1 **SRO Imp:** 2.8

Description: Ability to (a) predict the impacts of a Loss of Condensate Pump(s) on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of Condensate Pump(s).

Question:

Given the following plant conditions:

* Full Power Operations.
* All Plant Systems aligned for normal operation.
* Condensate Pumps 2P2A, 2P2B and 2P2C running.
* Condensate Pump 2P2A trips.

Which of the following automatic actions should occur?

A. Main Feed Water Pump Suction Cross-connect 2CV-0742 will auto close.

B. Main Feed Water Pump 2P1A will trip on low suction flow.

C. Recirculation valve for Condensate Pump 2P2A/C will auto open.

D. Condensate Pump 2P2D will auto start.

Answer:

D. Condensate Pump 2P2D will auto start.

Notes:

Answer "A" is incorrect since there is no auto close signal to the valve.
Answer "C" is incorrect since recirc valves are auto open on high pressure or low condensate flow.
Answer "B" is incorrect since low suction flow switch is normally bypassed.

References:

2106.016, Rev 36, Section 3.0 (Condensate and Feedwater Operations)
ANO-2-LP-AO-COND, Rev 09, Obj 6.0
STM 2-20, Rev 05, Section 2.3.3.2 (Condensate System)
STM 2-20, Rev 05, Section 3.5.5 (Condensate System)

Historical Comments:

08/22/01 Modified from QID 179 which was used on 1998 RO and SRO Exam

QID:	0359	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.2 to 41.9	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Main Feedwater System	**System Number**	059	**K/A:**	K1.05						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.1	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.2
Description:	Knowledge of the physical connections and/or cause effect relationships between the MFW System and the RCS.										

Question:

Given the following conditions:

* A reactor trip was initiated by low Steam Generator Pressure on both SGs ten minute ago.
* A Main Steam Isolation Signal, MSIS, was generated at the trip.
* Both SG levels are 23% Narrow Range and slowly restoring.
* RCS T-ave is 550°F.

The correct status of the following Main Feedwater System components would be:

A. Main Feedwater Pumps at 3030 rpm, Main Feed Regulating Valves Closed, Main Feed Regulating Bypass valves at approximately 19% open.

B. Main Feedwater Pumps at 3030 rpm, Main Feed Regulating Valves Closed, Main Feed Regulating Bypass valves at approximately 5% open.

C. Main Feedwater Pumps at turning gear speed, Main Feed Regulating Valves Closed, Main Feed Regulating Bypass valves at approximately 50% open.

D. Main Feedwater Pumps at turning gear speed, Main Feed Regulating Valves Closed, Main Feed Regulating Bypass valves at approximately 35% open.

Answer:

D. Main Feedwater Pumps at turning gear speed, Main Feed Regulating Valves Closed, Main Feed Regulating Bypass valves at approximately 35% open.

Notes:

The Main Feedwater Pumps will go to minimum speed of 3030 rpm on a reactor trip; however in this case both MSIVs should be closed due to an MSIS signal so no steam is available to the MFW turbine therefore they will slow down and go on the turning gear. This makes answers A and B wrong.

The MFRV always closes on a trip due to RTO. The MFRV Bypass valve modulate based on a T-ave of 548.24 at ~19% open position to a T-ave of 552 at 50% open. With the given conditions, T-ave should be calculated to be 550 which should place the bypass reg. valves at approximately 35 % open. This is based on a calculation of 4.12% flow demand at 550 degrees F T-ave. Therefore C is wrong.

References:

ANO-2-LP-RO-FWCD, Rev 04, Feedwater Control System, Objective 11
STM 2-69, Rev 7, Feedwater Control System, Section 2.1
STM 2-69, Rev 7, Feedwater Control System, FWCS Valve Demand vs. Flow Demand Graph
STM 2-19, Rev 7, Main Feedwater System, Section 7.0

Historical Comments:

QID:	0360	**Rev:**	001	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	AFW/Emergency Feedwater System	**System Number**	061	**K/A:**	A1.05						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.6	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.7

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including AFW flow/motor amps.

Question:

Given the following plant conditions:

* A plant shutdown and cooldown is in progress.
* Steam Generators are 60% and being fed with AFW Pump 2P75 only.

To prevent runnout on 2P75, maximum flow should not exceed:

A. 600 gpm

B. 800 gpm

C. 1000 gpm

D. 1200 gpm

Answer:

C. 1000 gpm

Notes:

Answer A is the limit for flow in a single 4 inch pipe train of AFW/EFW to on Steam Generator but is not a pump runnout limit. Answer D is the limit for two trains of AFW/EFW in a pipe train but is not a pump runnout limit. Answer B is not a EFW/AFW limit. Answer C is the correct limit listed in the precaution 5.17 of OP 2106.006, EFW System Operations

References:

ANO-2-LP-RO-EFW, Rev 05, Objective 3
2106.006, Rev. 053-01-0, EFW System Operations, Steps 5.10 and 5.17.

Historical Comments:

QID:	0361	**Rev:**	001	**Rev Date:**	10/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Aux/Emergency Feedwater	**System Number**	061	**K/A:**	K2.01						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.2	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.3
Description:	Knowledge of bus power supplies to the AFW/EFW System MOVs.										

Question:

Given the following plant conditions:

* Plant is operating at 100% power.
* 125 VDC Panel 2D26 supply breaker trips open due to a fault on the bus.
* 10 minutes later, the plant trips from 100% power due to a Loss of Offsite Power.
* Both steam Generator levels drop to 20% Narrow Range Level.
* All systems operate as designed under these conditions.

The response of the Emergency Feedwater System would be:

A. 2P-7A would automatically start and feed only the 'A' SG.

B. 2P-7B would automatically start and feed only the 'A' SG.

C. 2P-7A would automatically start and feed both SG's.

D. 2P-7B would automatically start and feed both SG's.

Answer:

D. 2P-7B would automatically start and feed both SG's.

Notes:

A loss of the 2D26 bus would prevent the normally closed upstream discharge MOVs from the 2P-7A EFW pump from opening and 2P7A would not start so no flow would come from the 2P-7A pump. This makes answers A and C wrong. Both AC powered MOVs on the discharge of the 2P-7B would come open providing a flow path through the downstream MOVs to provide flow to both steam generators and not just the 'A' SG. This makes answer B wrong.

References:

ANO-2-LP-RO-EFW, Rev 05, Objective 10
STM 2-19-2, Rev 10, EFW & AFW Systems, Section 2.3.3 and 2.3.3.2

Historical Comments:

9/4/01 -This question is a major modification of QID 11867 from the ANO-2 Operations Exam Bank.

QID:	0362	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.9	**Type:**	Radioactivity Release				
System Title:	Liquid Radwaste System	**System Number**	068	**K/A:**	A3.02						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	3.6	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to monitor automatic operation of the Liquid Radwaste System including automatic isolation.										

Question:

Given the following plant conditions:

* Unit 2 is at full power operation.
* A liquid release is in progress from Boric Acid Condensate Tank 2T-69A.
* Release Isolation 2CV-2330A is open, Release Isolation 2CV-2330B is closed.
* Control Room Release Handswitch 2HS-2333 on 2C14 is in position "RESET".
* Waste Condensate Tank (2T-21A) Pump 2P-53A Disch Valve (2CV-2122) is opened by WCO.

At this point, what would be the status of the 2T-69A release and the valve positions:

A. Release terminated, 2CV-2330A closed, 2CV-2122 closed.

B. Release terminated, 2CV-2330A closed, 2CV-2122 open.

C. Release in progress, 2CV-2330A open, 2CV-2122 closed.

D. Release in progress, 2CV 2330A open, 2CV-2122 open.

Answer:

B. Release terminated, 2CV-2330A closed, 2CV-2122 open.

Notes:

The interlock between a Boron Management System tank outlet valve and a Liquid Radwaste tank outlet valve will close release isolation 2CV-2330A to prevent inadvertent release of a non permitted tank. This makes answers C and D wrong. A is wrong because for the given conditions because the Waste Condensate Pump Discharge Valve will remain open.

References:

ANO-2-LP-WCO-BMS, Rev 07, Boron Management System, Objective 15
STM 2-52, Rev 05, LRW/BMS System Description, Sections 2.3.2.1, 3.5 and the LRW/BMS one line figures
E-2401, 2CV-2330 A and B electrical print.

Historical Comments:

QID:	0364	**Rev:**	001	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.9	**Type:**	Radioactivity Release				
System Title:	Waste Gas Disposal System	**System Number**	071	**K/A:**	K5.04						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.5	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.1

Description: Knowledge of the operational implications of the relationship of hydrogen/oxygen concentration to flammability as they apply to the Waste Gas Disposal System.

Question:

Which ONE of the following combination is an acceptable mix of hydrogen and oxygen in the inservice Gas Decay Tank (2T-18A/B/C) to prevent flammability? (Reference Material Provided)

A. 3% hydrogen and 11% oxygen

B. 5% hydrogen and 9% oxygen

C. 7% hydrogen and 7% oxygen

D. 9% hydrogen and 5% oxygen

Answer:

A. 3% hydrogen and 11% oxygen

Notes:

Refer to T.S Figure 3.11.-1 or 2104.022 Exhibit 1 or 2203.010 Attachment A which shows answer A as the only combination above in the acceptable region below the flammable region.

On a closed reference exam, provide Attachment A of 2203.010, H2/O2 Concentration graph showing acceptable, flammable and explosive mixture regions.

References:

ANO-2-LP-WCO-GRW, Rev 12, Gaseous Radwaste System, Objective 11
2203.010, Rev 007-01-0, H2/O2 Concentration High AOP, Attachment A
2104.022, Rev 032-03-0, Gaseous Radwaste System Operations, Exhibit 1
T.S. 3.11.3, Amendment 193, Explosive Gas Mixtures

Historical Comments:

1/17/2002- changed the difficulty of this question to a 2 from a 3 based on NRC feedback.

QID:	0365	**Rev:**	000	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**	43.5 & 43.3	**Section:**	3.7	**Type:**	Instrumentation				
System Title:	Radiation Monitoring System	**System Number**	072	**K/A:**	A2.02						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.8	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	2.9

Description: Ability to (a) predict the impacts of a detector failure on the Area Radiation Monitoring System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Question:

Given the following plant conditions:

* Unit 1 is in Mode 5 to repair a 20 gpm RCS leak on a loop drain valve packing.
* Unit 2 is in Mode 6 moving fuel from the Reactor to the Spent Fuel Pool.
* Both Control Room intake radiation monitors on Unit 1, 2RITS-8001A and B are inoperable.
* Unit 2 just declared both of their Control Room intake radiation monitors, 2 RITS-8750-1A and B inoperable.

The correct action to take for the above conditions would be to:(Reference Material Provided)

A. Continue normal activities, Control Room Intake Radiation Monitors are only required if either Unit is in Modes 1 through 4.

B. Place the Control Room Emergency Ventilation system in the recirculation mode of operation within one (1) hour.

C. Suspend the refueling shuffle inside Containment until BOTH Control Room Intake Radiation Monitors on Unit 2 are restored to operable.

D. Continue normal activities, have HP setup additional monitoring of Control Room envelope within four (4) hours.

Answer:

B. Place the Control Room Emergency Ventilation system in the recirculation mode of operation within one (1) hour.

Notes:

T.S. Table 3.3-6 Item 2b applies to the Control Room intake radiation monitors and based on Note 2 this TS applies any time handling of irradiated fuel is occurring and when either Unit is in Modes 1 through 4. This makes answers A, C and D wrong.

On a closed reference exam, provide 2104.007, Attachment B and Technical Specification 3.3.3.1

References:

ANO-2-LP-RO-RMON, Radiation Monitoring, Objective 18
2104.007, Rev 023-01-0, Attachment B, Note 3 and flowchart on Page 4 of 4
Technical Specification 3.3.3.1, Amendment 134, Action b.

Historical Comments:

1/17/2002. Added 2104.007 Attachment B and T.S. 3.3.3.1 as a reference for this question based on NRC feedback.

QID:	0366	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.7	**Type:**	Instrumentation				
System Title:	Radiation Monitoring System	**System Number**	072	**K/A:**	K5.01						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.7	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	

Description: Knowledge of the operational implications of radiation theory, including sources, types, units and effects as they apply to the Radiation Monitoring System

Question:

The N-16 Radiation Monitors 2RE-0200 and 2RE-0201 are gamma sensitive ___________________ type detectors and will provide valid Steam Generator tube leakrate calculations above ___________ percent power.

A. Geiger-Mueller; 10

B. Scintillation; 10

C. Geiger-Mueller; 20

D. Scintillation; 20

Answer:

D. Scintillation; 20

Notes:

The N-16 radiation monitors are scintillation type detectors so distracter A and C are wrong. Valid SG tube leakrates are only calculated above 20% power so distracter B is wrong.

References:

ANO-2-LP-RO-RMON, Rev 10, Radiation Monitoring, Objectives 6 and 21
STM 2-62, Rev 8, Radiation Monitoring System, Section 2.3.4

Historical Comments:

1/10/2002. Question was rewritten based on NRC feedback due to the GFES nature of the original question. BNC

QID:	0367	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.1	**10CFR55_43:**	43.2	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Reactor Coolant System	**System Number**	002	**K/A:**	2.1.32						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.4	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.8
Description:	Ability to explain and apply all system limits and precautions associated with the Reactor Coolant System.										

Question:

Given the following plant conditions:

* A plant cooldown is being conducted for an upcoming refueling outage.
* RCS pressure is 500 psia.
* RCS temperature is 250°F.

To prevent excessive thermal stresses from building up in the Reactor Vessel, the rate of cooldown should be limited to:

A. Less than 25°F per hour or 12.5°F in any half hour period.

B. Less than 60°F per hour or 30°F in any half hour period.

C. Less than 80°F per hour or 40°F in any half hour period.

D. Less than 100°F per hour or 50°F in any half hour period.

Answer:

D. Less than 100°F per hour or 50°F in any half hour period.

Notes:

D is the correct answer in accordance with T.S 3.4.9.1

References:

ANO-2-LP-RO-RCS, Rev 12, Objective 27.c
2102.010, Rev 033-02-0, Plant Cooldown, Step 5.1
Technical Specification 3.4.9.1, Amendment 124

Historical Comments:

QID:	0368	**Rev:**	000	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.2	**Type:**	RCS Inventory Control				
System Title:	Emergency Core Cooling System	**System Number**	006	**K/A:**	A4.03						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.5	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.5

Description: Ability to operate and/or monitor in the control room transfer from boron storage tank to boron injection tank.

Question:

Given the following conditions:

* Plant is at 100% power.
* Safety Injection Tank (SIT) "A" is at 80.3% and 605 psig.
* CRS directs CBOT to fill SIT "A" using HPSI Pump 2P-89A from the RWT.
* After filling, SIT "A" is at 87.7% and 625 psig.

In accordance with Technical Specifications (TS), after completion of this task the CBOT should:

A. Contact Chemistry to sample SIT "A" within 6 hours.

B. Restore SIT "A" level to within the TS limits

C. Restore SIT "A" pressure to within the TS limits.

D. Take no additional actions based on this task.

Answer:

C. Restore SIT "A" pressure to within the TS limits.

Notes:

Safety Injection Tanks are required to be sampled if a > 5% level rise occurs unless the level rise occurred due to an addition from the RWT. This is the source of water for the HPSI pump at 100% power. This makes distracter A wrong. SIT A level parameters given are within the TS limits therefore distracter B is wrong. Answer C is correct because the SIT pressure is above the TS limit of 624. Distracter D is wrong because an action is required.

References:

ANO-2-LP-WCO-SIT, Rev 07, Safety Injection Tanks, Objectives 3 and 10
STM 2-05, Rev 09, ECCS, Section 5.1
2104.001, Rev 025-04-0, SIT Operations, Steps 5.1 and 5.3
Technical Specification 3.5.1, Amendment 192

Historical Comments:

QID:	0369	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.3	**Type:**	RCS Pressure Control				
System Title:	Pressurizer Pressure Control System	**System Number**	010	**K/A:**	K2.02						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of bus power supplies to the controller for PZR spray valve.										

Question:

Which ONE of the following buses supplies power to Pressurizer Pressure Controller Channel A, 2PIC-4626A?

A. 2Y1

B. 2Y2

C. 2Y3

D. 2Y4

Answer:

A. 2Y1

Notes:

2Y-1 is the 120VAC electrical power supply to the 2CO4, Reactor Coolant system panel for the A train components. 2Y2 is for the B train components. 2Y3 and 2Y4 are additional buses that were installed after plant startup to power up the Diverse Scram System and Fire Water Panels.

References:

ANO-2-LP-RO-PZR, Objective 2
STM 2-3-1, Pressurizer Pressure and Level Control, Section 2.4
STM 2.32-4, 120 VAC Distribution System, Section 2.1
OP 2107.003, Inverter and 120 VAC Electrical System Operation, Exhibit 1
Electrical Print E-2704

Historical Comments:

QID:	0370	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	AP	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.3	**Type:**	RCS Pressure Control				
System Title:	Pressurizer Pressure Control System	**System Number**	010	**K/A:**	K6.03						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.2	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.6
Description:	Knowledge of the effect of a loss or malfunction of the PZR sprays and heaters will have on the PZR Pressure Control System.										

Question:

Given the following:

* The plant is at 100% Power.
* Pressurizer Pressure Channel Select Switch is selected to "A".
* Pressurizer Pressure Controller, 2PIC-4626A, is in Automatic.
* All other Pressurizer pressure controls are in their normal configuration.
* Selected pressure transmitter 2PT-4626A fails High.
* All other systems respond as designed.
* No operator action is taken.

Considering only the effects of Pressurizer pressure, which ONE of the following correctly describes the response of the plant?

A. All backup heaters energize, spray valves stay closed, and the plant trips on RPS trip on high Pressurizer Pressure at 2362 psia.

B. All backup heaters energize, spray valves stay closed, and the plant trips on CPC Aux trip on high Pressurizer Pressure at 2375 psia.

C. All backup heaters remain off, spray valves open, and the plant trips on RPS trip on Low Pressurizer Pressure at 1675 psia.

D. All backup heaters remain off, spray valves open, and the plant trips on CPC Aux trip on Low Pressurizer Pressure at 1860 psia.

Answer:

D. All backup heaters remain off, spray valves open, and the plant trips on CPC Aux trip on Low Pressurizer Pressure at 1860 psia.

Notes:

With the pressure transmitter failed high, the output of the controller will sense a high input pressure and open the main spray valves fully and de-energize all backup heaters. With the spray valves fully open, Pressurizer Pressure will drop with no operator action and trip the plant on a CPC auxiliary trip pressure of 1860 psia.

References:

ANO-2-LP-RO-PZR, Rev 2, Objectives 2, 3, 4, and 5.
STM 2-3-1, Rev 5, Pressurizer Pressure and Level Control, Sections 2.2.4, 2.2.6, and the figure on page 33
T.S. Table 2.2-1, Amendment 222, RPS Trip setpoints
2105.001, Rev 023-06-0, CPC/CEAC Operations, Section 6.1.4 and 6.1.5

Historical Comments:

QID:	0371	**Rev:**	001	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	NRC Exam Bank	**Originator:**	Hatman		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.2	**Type:**	RCS Inventory Control				
System Title:	Pressurizer Level Control System	**System Number**	011	**K/A:**	K2.02						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.1	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.2
Description:	Knowledge of bus power supplies to the PZR heaters.										

Question:

Which of the following pressurizer heater banks are powered by 480V Vital Power?

A. Proportional heaters banks #1 and #2, backup heaters #1 and #2.

B. Backup heaters banks #1, #2, #3, #4.

C. Proportional heater banks #1 and #2.

D. Backup heaters banks #1 and #2.

Answer:

C. Proportional heater banks #1 and #2.

Notes:

Backup Heater Banks are non vital 480 volt AC powered which makes answers A, B and D wrong.

References:

2107.002, Rev 015-01-0, Attachment B, (ESF Electrical System Operation)
STM2-3-1, Rev 5, Section 2.4 Table (Pressurizer Pressure and Level Control)
ANO-2-LP-RO-RCS, Rev 12, Objective 10.0
ANO-2-LP-RO-PZR, Rev 2, Objective 3

Historical Comments:

Rev 001 - 08/11/98 - Added procedure 2107.002 to reference list per NRC review comment. This question was used on the 1998 NRC Exam for the ROs only

QID:	0372	**Rev:**	000	**Rev Date:**	09/17/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.7	**Type:**	Instrumentation				
System Title:	Reactor Protection System	**System Number**	012	**K/A:**	K6.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.8	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.3
Description:	Knowledge of the effect of a loss or malfunction of bistables and bistable test equipment will have on the RPS.										

Question:

Given the following plant conditions:

* Plant is operating at 100% power.
* PPS Channel B High Containment Pressure bistable is in the tripped condition and bypassed.
* PPS Channel C High Containment Pressure transmitter fails high.
* The CBOT goes to bypass affected tripped bistable on PPS Channel C.

Which ONE of the following is the correct response of the PPS System on the High Containment Pressure bistables when the CBOT goes to bypass the affected tripped bistable on Channel C PPS?

A. Both the Channel B and C bistables will be bypassed.

B. Both the Channel B and C bistables will NOT be bypassed.

C. The C bistable will be bypassed and the B bistable will NOT be bypassed.

D. The B bistable will be bypassed and the C bistable will NOT be bypassed.

Answer:

B. Both the Channel B and C Bistables will NOT be bypassed.

Notes:

Due to the design of the PPS, it will not physically allow two channels to be placed in bypass at the same time. There are conditions where a trip in one channel is placed in "trip" and the same trip in another channel is placed in "bypass". This would change the normal trip logic from 2 out of 4 to 1 out of 2. Distracters C & D appear to be valid distracters by presenting incorrect information.

References:

STM 2-63, Rev 05, Reactor Protection System, Section 6.2.1
ANO-2-LP-RO-RPS, Rev 07, Reactor Protection System, Objective 10.a

Historical Comments:

QID:	0373	**Rev:**	000	**Rev Date:**	09/18/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.1	**Type:**	Reactivity Control				
System Title:	Rod Position Indication System	**System Number**	014	**K/A:**	K5.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.7	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.0

Description: Knowledge of the operational implications of the reasons for differences between Rod Position Indication System and step counter.

Question:

Control Element Assemblies (CEAs) have two means of position indication, Reed Switch Position Transmitters (RSPT) and Pulse Counters.

Which ONE of the following correctly describes where the RSPT and Pulse Counter output signals are sent? (PDIL = Power Dependent Insertion Limit, CPC = Core Protection Calculators, COLSS = Core Operating Limits Supervisory System, UEL = Upper Electrical Limit, LEL = Lower Electrical Limit)

	RSPT	Pulse Counter
A.	CPC Dropped Rod Contact UEL, LEL Lights	COLSS PDIL Alarms Major CEA Deviation Alarms
B.	CPC PDIL Alarms Major CEA Deviation Alarms	COLSS Dropped Rod Contact UEL, LEL Lights
C.	COLSS Dropped Rod Contact UEL, LEL Lights	CPC PDIL Alarms Major CEA Deviation Alarms
D.	COLSS PDIL Alarms Major CEA Deviation Alarms	CPC Dropped Rod Contact UEL, LEL Lights

Answer:

A.	CPC Dropped Rod Contact UEL, LEL Lights	COLSS PDIL Alarms Major CEA Deviation Alarms

Notes:

The RSPTs are the safety related rod position indications which go to the safety systems and the maximum rod stop indications such as CPCs, Upper Electrical Limit (UEL), Lower Electrical Limit (LEL) and the dropped rod contacts. The Pulse counters are the non-safety rod position indication and go to the non-safety systems such as COLSS and the Plant Computer which drives the Power Dependent Insertion Limit (PDIL) Alarm and the CEA Group deviation Alarms. Answer A is the only one above with the correct combination.

References:

ANO-2-LP-RO-CEDM Rev 8, Objective 3
STM 2-02 Rev 7, Control Element Drive Mechanism Control System, Sections 3.7, 4.2.1.6, and 9.1.1

Historical Comments:

QID:	0374	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.2 to 41.5	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Containment Spray System	**System Number**	026	**K/A:**	K1.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	4.2	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	4.2

Description: Knowledge of the physical connections and/or cause-effect relationships between the Containment Spray System and the ECCS.

Question:

Given the following plant conditions:

* A Large Break LOCA is in progress.
* Containment Pressure is 49 psig.
* Containment Temperature is 247°F.
* The RWT level is 5.4%.
* All ECCS components operate as designed.

At this point in the accident, the system that is providing long term cooling for the core is:

A. The High Pressure Safety Injection Pumps using water from the RWT.

B. Low Pressure Safety Injection Pumps through the SDC Heat Exchangers.

C. The Containment Spray Pumps through the SDC Heat Exchangers.

D. All available Charging Pumps and the Letdown Heat exchanger.

Answer:

C. The Containment Spray Pumps through the SDC Heat Exchangers.

Notes:

A is wrong because the HPSI will have a suction path from the containment sump. B is wrong because the LPSI Pump is not aligned to the SDC heat exchanger and it trips on a RAS at 6% in the RWT. D is wrong because the Charging pump capacity is not sufficient to remove long term decay heat buildup in the core.

References:

ANO-2-LP-RO-SPRAY, Containment Spray System, Objective 9
STM-2-08, Containment Spray System, Revision 7, Sections 1.0, 2.1, 3.5, 4.2 and 5.2
Technical Specification 3.6.2.1 Bases

Historical Comments:

QID:	0377	**Rev:**	000	**Rev Date:**	09/06/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.2 & 43.5	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Steam Generator	**System Number**	035	**K/A:**	2.1.12						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.9	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	4.0
Description:	Ability to apply technical specifications for the Steam Generator System.										

Question:

Given the following plant conditions:

* The plant is in mode 5 coming out of a refueling outage.
* Re-pressurization of the RCS is in progress in preparation for a plant heatup.
* RCS temperature is being maintained by the in-service SDC train at 88°F.
* RCS Pressure is raised to 295 psig.

Assuming RCS temperature remains the same, RCS pressure should be reduced below _______ psig within ______ minutes to comply with the Steam Generator Pressure/Temperature Limitation LCO.

A. 275; 60

B. 275; 30

C. 285; 60

D. 285; 30

Answer:

B. 275; 30

Notes:

B is the correct answer based on T.S LCO 3.7.2.1

References:

ANO-2-LP-RO-STEAM, Revision 11, Objective 1L
Technical Specification 3.7.2.1

Historical Comments:

QID:	0378	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Main and Reheat Steam System	**System Number**	039	**K/A:**	A3.02						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.1	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.5

Description: Ability to monitor automatic operation of the Main/Reheat Steam System, including isolation of the M/R Steam System.

Question:

Given the following plant conditions:

* The plant has tripped from 100% power due to a Loss of Offsite Power (LOOP).
* RCS pressure is 2100 psia and slowly dropping.
* Steam Generator pressure is 940 psia and slowly dropping in both SGs.
* The crew has diagnosed a LOOP and entered EOP ORP 2202.007.

Which ONE of the following actions should be taken to prevent further pressure reduction in the Steam Generators and overcooling of the RCS.

A. Close the Upstream ADV isolation because the Upstream ADV fails open on a LOOP.

B. Close the MSIVs because the Downstream ADV fails open on a LOOP.

C. Close the MSIVs because the MSR 1st stage reheat isolations remain open on a LOOP.

D. Close the MSIVs because the MSR 2nd stage reheat isolations remain open on a LOOP.

Answer:

D. Close the MSIVs because the MSR 2nd stage reheat isolations fail open on a LOOP.

Notes:

A is incorrect, the Upstream ADVs do fail open on a Loss of Power but are normally isolated at power by a closed MOV. B is incorrect because the downstream ADVs are powered from Vital buses and would retain power and control at setpoint. C is incorrect because the MSR 1st stage reheat is supplied from turbine extraction steam which would be isolated by the Main Turbine Stop valves automatically. D is correct because the MSR 2nd stage reheat isolation come directly off of the Main Steam Header and is a Non Vital Powered MOV.

References:

ANO-2-LP-RO-STEAM, Revision 11, Objective 3h
STM 2-16, Revision 4, Section 3.3.1.2 and figure on page 17
STM 2-15, Revision 6, Section 3.2.8 and figure on page 59
OP 2202.007, LOOP EOP, Revision 5, Step 5
EOP 2202.007, LOOP EOP Technical Guide, Revision 5, Step 5

Historical Comments:

QID:	0379	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	k	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Condenser Air Removal System	**System Number**	055	**K/A:**	K3.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	2.7

Description: Knowledge of the effect that a loss or malfunction of the Condenser Air Removal System will have on the main condenser.

Question:

Given the following plant conditions:

* The plant is at 100% power.
* Condenser Vacuum Pump 2C-5A is running, 2C-5B is in standby.
* Condenser Vacuum starts to degrade.

Which ONE of the following describes the design of the Condenser Vacuum System that will prevent continued degradation of Condenser vacuum.

A. Condenser Vacuum Pump 2C-5A will shift to the "Hogging" mode of operation at 4" HgA vacuum in the condenser and prevent further loss of vacuum.

B. Condenser Vacuum Pump 2C-5A will shift to the "Holding" mode of operation at 6" HgA vacuum in the condenser and prevent further loss of vacuum.

C. Condenser Vacuum Pump 2C-5B will startup at 4" HgA vacuum in the condenser and will be operating in the "Holding" mode after the inlet diaphragm valves are opened.

D. Condenser Vacuum Pump 2C-5B will startup at 6" HgA vacuum in the condenser and will be operating in the "Hogging" mode after the inlet diaphragm valves are opened.

Answer:

C. Condenser Vacuum Pump 2C-5B will startup at 4" HgA vacuum in the condenser and will be operating in the "Holding" mode after the inlet diaphragm valves are opened.

Notes:

The "Hogging" mode of operation on the operating or the started standby air evacuation pump will not occur until vacuum degrades to 7" HgA. This makes answers A and D wrong. The "Holding" mode of operation is achieved on an increasing vacuum at 5"HgA. For the given conditions the 2C-5A Vacuum pump would already be in the holding mode of operation; therefore, answer B is wrong. Answer D is also wrong because the auto start of the standby pump is at 4" HgA vacuum.

References:

ANO-2-LP-AO-VACUM, Revision 8, Objectives 3, 11 and 14
STM 2-22, Revision 6, Sections 2.3, 3.1,3.2 and 3.3
OP 2106.010, Revision 015-05-0, Section 3.0

OP 2203.012C, Revision 021-00-0, ACA for Vacuum Pump 2C-5B Auto Start -2K03 E-4

Historical Comments:

QID:	0381	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.6	**Type:**	Electrical				
System Title:	DC Electrical Distribution	**System Number**	063	**K/A:**	K3.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.7	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the EDG.										

Question:

Given the following plant conditions:

* The Plant is at 100% power.
* 2DG2 is at 2850 KW for surveillance testing.
* A fault on 2D02 causes the loss of all Green DC.
* All other equipment operates as designed.
* Assume no operator action.

One minute into this event, 2DG2 would be:

A. Running and paralleled with offsite at approximately 2850 KW

B. Secured due to a loss of electrical governor control power

C. Running but unloaded because the output breaker opened

D. Running on the mechanical governor and loaded to 2A4 only

Answer:

A. Running and paralleled with offsite at approximately 2850 KW

Notes:

To start and stop an EDG requires in-house DC control power but when the EDG is running, the power to the governor speed control circuit is supplied from potential transformers on the output voltage of the Diesel itself. Thus when the diesel is running a loss of its associated vital DC should have no impact and the Diesel will remain tied to offsite at full load. No Breakers should reposition because they have lost control power to open them so answer B and C are wrong. Answer A is wrong because it takes DC power to energize a shutdown solenoid to secure the EDG.

References:

ANO-2-LP-RO-EDG, Revision 8, Objective 3
STM 2-31, EDGs, Revision 9, Sections 1.1 and 2.4.7

Historical Comments:

From in-house bank QID 7880

QID:	0382	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.6	**Type:**	Electrical				
System Title:	Emergency Diesel Generator System	**System Number**	064	**K/A:**	A3.07						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.6	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.7
Description:	Ability to monitor automatic operation of the EDG system, including load sequencing.										

Question:

Given the following plant conditions:

* A Plant trip has occurred due to a loss of offsite power.
* Pressurizer Pressure is 1550 psia and dropping.
* Both EDGs start and their output breakers close as designed.

Which ONE of the following list the major pump starts on the safety busses in the correct order beginning with the first pump start?

A. HPSI pumps, Charging Pumps, Service Water Pumps, LPSI Pumps.

B. Charging Pumps, HPSI Pumps, LPSI Pumps, Service Water Pumps.

C. LPSI Pumps, Service Water Pumps, HPSI Pumps, Charging Pumps.

D. Service Water Pumps, HPSI Pumps, LPSI Pumps, Charging Pumps.

Answer:

D. Service Water Pumps, HPSI Pumps, LPSI Pumps, Charging Pumps.

Notes:

Answer D list the correct order in accordance with the Diesel Loading Table. This make answers A, B, and C incorrect.

References:

ANO-2-LP-RO-EDG, Revision 8, Objective 3
STM 2-31, EDG System Description, Revision 8, Diesel Load Table

Historical Comments:

QID:	0383	**Rev:**	000	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.7	**Type:**	Instrumentation				
System Title:	Process Radiation Monitoring System	**System Number**	073	**K/A:**	K5.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.0

Description: Knowledge of the operational implications of radiation theory, including sources, types, units and effects as they apply to the Process Radiation Monitoring System.

Question:

Given the following plant conditions:

* Plant has returned to 100% power from 70% power.
* Annunciator 2K12-A1, "LETDOWN RADIATION HI/LO has actuated.
* CBOT is directed to monitor RCS Gross and Iodine activities on Letdown Radmonitor Recorder, 2RR-4806, on 2C-22.

If RCS Iodine 131 Activity has caused the alarm, then ___________________ should be suspected but if RCS Gross Activity has caused the alarm, then __________________ should be suspected.

A. RCS crud burst; Letdown filter damage

B. Fuel cladding damage; RCS crud burst

C. Letdown filter damage; Fuel cladding damage

D. RCS crud burst; Fuel cladding damage

Answer:

B. Fuel cladding damage; RCS crud burst

Notes:

The differential pressure across the Letdown radiation monitors is driven by the pressure drop across the Letdown filter. The only way Letdown filter damage could cause a rise in RCS activity is if it as located upstream of the radiation monitor. As such they are in parallel to the radiation monitors thus answers A and C are wrong. D is wrong because it is the reverse of the correct answer B.

References:

ANO-2-LP-RO-RMON, Revision 10, Objective 19
STM 2-04, CVCS, Revision 12, Section 2.1.3 and the figure on page 62
STM 2-62, Radiation Monitoring System, Revision 6, Section 2.2.1
OP-2203.012L, ACA for Letdown Radiation HI/LO, Revision 030-01-0 Window A-1
OP-2203.020, High RCS Activity, Revision 007-05-0, Step 7 and its associated Technical Guide Step

Historical Comments:

QID:	0384	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.8	**Type:**	Plant Service Systems				
System Title:	Circulating Water System	**System Number**	075	**K/A:**	K4.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of Circulating Water System design feature(s) and interlock(s) which provide for heat sink.										

Question:

Given the following plant conditions:

* Plant is at 100 % power.
* Circulating Water Pump 2P-3B trips on undervoltage due to an electrical fault.
* All equipment operates as designed.

Vacuum will degrade and:

A. cause a turbine trip because the discharge valve will not close until the Circulating Water System is shutdown allowing vacuum to drop below 7.8" HgA vacuum .

B. cause a Turbine trip because the discharge valve closes completely in slow speed allowing vacuum to drop below 7.8" HgA vacuum.

C. then stabilize at a degraded pressure because the discharge valve starts closing in fast speed and then shifts to slow speed as the valve approaches the closed seat.

D. then stabilize at a degraded pressure because the discharge valve closes completely in fast speed preventing backflow through the idle pump.

Answer:

C. then stabilize at a degraded pressure because the discharge valve starts closing in fast speed and then shifts to slow speed as the valve approaches the closed seat.

Notes:

The closing actuation of the CW pump discharge valves actuates two different speed clutches. 1st the fast speed clutch is engaged on a pump trip to rapidly ramp the valve closed to 17% open to prevent reverse flow and thus a loss of heat sink to the condenser. After this the slow speed clutch is engaged to continue closing the valve against a higher differential pressure until it is fully closed. This interlock is designed to prevent a turbine trip on low vacuum from 100% power due to a loss of the condenser heat sink on a loss of one Circulating Water Pump.

References:

ANO-2-LP-AO-CIRC, Revision 7, Objective 3
STM 2-40-1, Circulating Water System, Revision 9, Section 2.4

Historical Comments:

From Site Exam Bank

QID: 0385 **Rev:** 000 **Rev Date:** 01/10/2002 **RO Select:** Yes **SRO Select:** Yes **Points:** 1.00

Lic Level: RS **Difficulty:** 2 **Taxonomy:** K **Source:** New **Originator:** Coble

10CFR55_41: 41.5 **10CFR55_43:** 43.5 **Section:** 3.6 **Type:** Electrical

System Title: Emergency Diesel Generator System **System Number** 064 **K/A:** A2.09

RO Tier: 2 **RO Group:** 2 **RO Imp:** 3.1 **SRO Tier:** 2 **SRO Group:** 3 **SRO Imp:** 3.3

Description: Ability to predict the impact of a malfunction of the EDG System during synchronizing an EDG with other electric power supplies and based on those predictions, use procedures to correct, control or mitigate the consequences of the malfunction or operation.

Question:

Given the following plant conditions:

* The plant is at 100% power.
* 2DG1 is in progress of being paralleled with offsite power in accordance with 2104.036, EDG Operations.
* All systems operate as designed.

In order to guarantee that reactive load will be "OUT" or "POSITIVE" when the output breaker is closed, the procedure directs that the 2DG1 voltage on 2C-33 should be ___________________ than System (Running) voltage and 2DG1 voltage as indicated on SPDS should be __________ than System (Running) voltage.

A. Approximately 100 volts higher; higher

B. Approximately 100 volts lower: lower

C. Approximately 50 volts higher; 20 volts higher

D. Approximately 50 volts lower; 20 volts lower

Answer:

A. Approximately 100 volts higher; higher

Notes:

This answer is based on the guidance found in the referenced procedure. To have outgoing or positive VARs incoming or EDG voltage must be higher than running or system voltage. This makes answers B and D wrong. C is wrong because the corrective action of the CR directs approximately 100 volts higher and just higher when obtaining a diverse indication such as SPDS. These are conditions found in the guidance of the procedure.

References:

ANO-2-LP-RO-EDG, Revision 8, Objective 4
OP 2104.036, EDG Operations, Revision 045-03-0, Steps 9.12.2 and 11.21.2.
Condition Reports CR-ANO-2-2001-0158 and 0491

Historical Comments:

1/10/2002 Added the name of the procedure to the stem for 2104.036 based on NRC feedback. BNC

QID:	0386	**Rev:**	000	**Rev Date:**	10/08/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.8	**Type:**	Plant Service Systems				
System Title:	Fire Protection System	**System Number**	086	**K/A:**	A4.01						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	3.3	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to manually operate and/or monitor in the control room the fire detection panels.										

Question:

Which ONE of the following Fire Protection Deluge Sprinkler Systems can be manually actuated from the Control Room Fire Protection/Detection Control Panel 2C-343.

A. MFWP Lube Oil Reservoir

B. Hydrogen Seal Oil Skid

C. Unit Auxiliary Transformer

D. EDG Fuel Oil Storage Vaults

Answer:

D. EDG Fuel Oil Storage Vaults

Notes:

D is the correct answer based on the controls located on the Control Panel 2C-343. Answers A, B, and C are deluge sprinkler systems with automatic and local actuation controls.

References:

ANO-2-LP-RO-FPROT, Revision 9, Objective 4
STM 2-60, Fire Protection System, Revision 4, Section 7.0 and Table 4

Historical Comments:

QID:	0387	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	An	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Residual Heat Removal System	**System Number**	005	**K/A:**	K5.02						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.4	**SRO Tier:**	2	**SRO Group:**	3	**SRO Imp:**	3.5

Description: Knowledge of the operational implications of the need for adequate subcooling as they apply to the Residual Heat Removal System.

Question:

Given the following plant conditions:

* The plant has shutdown for a refueling outage.
* Core outlet temperature is 211°F.
* RCS Pressure is 50 psia.
* Pressurizer level is 40% with a steam bubble present.
* All RCPs are secured.
* Steam Generator Levels are 60% Narrow range being maintained with the AFW Pump 2P-75.
* All systems are currently operable.
* No dilution of the RCS boron is planned.
* Red Train Shutdown Cooling (SDC) Loop is in service.
* Green Train SDC Loop is in the process of being aligned to SDC from the injection mode.
* RCS heatup rate based on decay heat with no SDC heat removal is 3°F/minute.

If the Red Train SDC loop is secured, to ensure compliance with the Technical Specification LCO one train of SDC should be returned to service within?

A. 60 minutes

B. 45 minutes

C. 32 minutes

D. 20 minutes

Answer:

D. 20 minutes

Notes:

The RCS is required to be maintained 10 degrees below saturation temperature with no SDC or RCPs in operation by the asterisked note in T.S. 3.4.1.3. In this case RCS saturation temperature is 281°F based on 50 psia. 281°F -10°F = 271°F. 271°F - 211°F = 60°F. 60°F/3°F/minute = 20 minutes.

Both RCS loops are operable based on SG levels, available feed source, and an operable SDBCS.

References:

ANO-2-LP-RO-SDC, Revision 5, Objective 4
ANO-2-LP-SRO-TS, Revision 7, Objective 14
Technical Specification 3.4.1.3
1015.008, Unit 2 SDC Control, Revision 017-02-0, Step 5.2

Historical Comments:

QID:	0388	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.2 to 41.9	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Pressurizer Relief/Quench Tank System	**System Number**	007	**K/A:**	K1.03						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.0	**SRO Tier:**	2	**SRO Group:**	3	**SRO Imp:**	3.2
Description:	Knowledge of the physical connections and/or cause-effect relationships between the Quench Tank System and the RCS.										

Question:

Which ONE of the following flow paths, when in service, could be directed to the Quench Tank (2T-42)?

A. RCP 4th stage vapor seal drains

B. Reactor vessel head vents

C. Reactor vessel head inner gasket leakoff

D. Reactor coolant loop drains

Answer:

B. Reactor vessel head vents

Notes:

The answers given in A, C and D are all directed to the Reactor Drain Tank making them wrong.

References:

ANO-2-LP-RO-RCS, Revision 12, Objective 25
STM 2-3, RCS, Revision 8, Section 2.3 and 2.4
STM 2-52, Liquid Radwaste/Boron Management, Revision 5, Section 3.1

Historical Comments:

From site exam bank

QID:	0389	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Hydrogen Recombiner and Purge Control Syst	**System Number**	028	**K/A:**	A4.01						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	4.0	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	4.0

Description: Ability to manually operate and/or monitor in the control room the Hydrogen Recombiner and Purge Control System controls.

Question:

Given the following plant conditions:

* A large break LOCA has occurred inside Containment.
* Containment Hydrogen concentration is 2.2%.
* The CRS has directed the CBOT to start both Hydrogen Recombiners.

To ensure proper Hydrogen Recombiner operation after the startup, do not exceed a maximum Recombiner output power of ________ KW and a Recombiner heater corrected outlet temperature of _________ °F.

A. 25; 1400

B. 75; 1000

C. 25; 1000

D. 75; 1400

Answer:

D. 75; 1400

Notes:

75 KW and 1400°F are the maximum allowed limits imposed by the Hydrogen Recombiner vendor to prevent damage to the units during operation. 1000°F is below the procedural guided minimum limit to maintain on the heater output to ensure actual recombination.

References:

ANO-2-LP-RO-CONH2, Revision 1, Objectives 14 and 15
OP 2104.044, Containment Hydrogen Control Operations, Revision 025-05-0, Steps 5.3,5.4, 9.16, and 9.18.
STM 2-6, Containment Combustible Gas Control, Revision 5, Sections 3.3 and 4.1.1.

Historical Comments:

QID:	0390	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.8	**Type:**	Plant Service Systems				
System Title:	Fuel Handling Equipment System	**System Number**	034	**K/A:**	A4.02						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.5	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.9

Description: Ability to manually operate and/or monitor neutron levels in the control room during refueling operations.

Question:

Given the following plant condition:

* The plant is in Mode 6 with a refueling shuffle in progress from the core to the Spent Fuel Pool.

To allow continued fuel movement, _______ source range neutron flux monitor(s) must be operating with visual indication in the control room and ______ source range neutron flux monitor(s) must be operating with audible indication in the containment and control room.

A. 1; 1

B. 2; 1

C. 1; 2

D. 2; 2

Answer:

B. 2; 1

Notes:

Technical Specification 3.9.2 requires 2 visual and 1 audible indication of neutron flux levels available to the control room crew during core alterations.

References:

ANO-2-LP-RO-NI, Revision 9 Objectives 5 and 8
Technical Specification 3.9.2 LCO
op 2502.001, Refueling Shuffle, Revision 029-04-0, Step 6.19

Historical Comments:

QID:	0391	**Rev:**	000	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Steam Dump System and Turbine Bypass Con	**System Number**	041	**K/A:**	K3.01						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.2	**SRO Tier:**	2	**SRO Group:**	3	**SRO Imp:**	3.3
Description:	Knowledge of the effect that a loss or malfunction of the Steam Dump System will have on SGs.										

Question:

Given the following plant conditions:

* The plant has experienced a Loss of all Off Site Power from 100% power.
* #2 EDG has failed to start.
* #1 EDG has started and tied onto 2A3.
* Instrument Air header pressure is 0 psig.
* 2202.007, Loss of Off Site Power has been entered.
* Both MSIVs have been closed.
* Steam Generator pressures are 1070 psia each controlling on Main Steam Safeties.

Which ONE of the following actions could be taken from the control room to restore Steam Generator pressure control to the normal shutdown operating band of 950 to 1050 psia?

A. Control the A SG pressure using the A SG Upstream Atmospheric Dump Valve, 2CV-1001.

B. Control the B SG pressure using the B SG Upstream Atmospheric Dump Valve, 2CV-1051.

C. Control the A SG pressure using the A SG Upstream Atmospheric Dump Valve Isolation Valve, 2CV-1002.

D. Control the B SG pressure using the B SG Upstream Atmospheric Dump Valve Isolation Valve, 2CV-1052.

Answer:

D. Control the B SG pressure using the B SG Upstream Atmospheric Dump Valve Isolation Valve, 2CV-1052.

Notes:

Both SG upstream Atmospheric Dump Valves (ADVs) fail open on loss of instrument air so they would not be available for pressure control. The power supply to the A SG upstream ADV MOV isolation is powered from the Green or B train of vital power so it is unavailable. The power supply to the B SG upstream ADV MOV isolation is powered from the Red or A train of vital power so it is available to modulate open and closed from the control room handswitch.

References:

ANO-2-LP-RO-SDBCS, Revision 9, Objective 16
ANO-2-LP-RO-STEAM, Revision 11, Objective 2m
STM 2-23, SDBCS, Revision 6, Section 2.12
STM 3.2.3, Main Steam and SG System, Revision 6, Section 3.2.3 and figure on page 59
OP 2202.007, LOOP ORP, Revision 5, Step 22

Historical Comments:

From in house bank

QID:	0392	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**	43.5	**Section:**	3.4	**Type:**	RCS Heat Removal				
System Title:	Main Turbine Generator System	**System Number**	045	**K/A:**	A2.08						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	2.8	**SRO Tier:**	2	**SRO Group:**	3	**SRO Imp:**	3.1

Description: Ability to (a) predict the impacts of steam dumps not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary) on MTG system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

Question:

Given the following plant conditions:

* The plant is at 100% power.
* A failure in the Steam Dump and Bypass Control System (SDBCS) has caused the first valve in the Valve Group Sequence Program, Steam Dump to the Main Condenser, 2CV-0303, to fail full open.
* All other SDBCS valves remain closed.

If no operator action was taken, Reactor power would rise to _________ and the correct action to take to close 2CV-0303 would be to ____.

A. 110%; place the permissive handswitch for 2CV-0303 to MANUAL position

B. 105%; place the permissive handswitch for 2CV-0303 to OFF position

C. 110%; place the permissive handswitch for 2CV-0303 to OFF position

D. 105%; place the permissive handswitch for 2CV-0303 to MANUAL position

Answer:

B. 105%; place the permissive handswitch for 2CV-0303 to OFF position

Notes:

The capacity of 2CV-0303 is 5% and the other steam dumps are 11.5%. One of the other dumps would go above the Linear Power high trip setpoint of 110% power which would be the maximum power achieved. This makes answers A and C wrong. Going to Manual on the permissive switch provides a one out of two required signals to open 2CV-0303. Going to Off prevents any open signals from reaching 2CV-0303, removes all air and the valve fails closed.

References:

ANO-2-LP-RO-SDBCS, Revision 9, Objectives 1 and 9.
STM 2-23, SDBCS, Revision 6, Sections 1.2, 2.1.10, and 2.3.4.
OP 2105.008, SDBCS Operations, Revision 014-01-0, Section 3.0 and step 9.4

Historical Comments:

QID:	0393	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.8	**Type:**	Plant Service Systems				
System Title:	Instrument Air System	**System Number**	078	**K/A:**	A3.01						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.1	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to monitor automatic operation of the Instrument Air System including air pressure.										

Question:

Given the following plant conditions:

* Instrument Air Compressor 2C-27A is the LEAD compressor and running unloaded.
* Instrument Air Compressor 2C-27B is the LAG compressor and is in standby after cycling off on low Instrument Air load.
* Instrument Air Pressure at the outlet of the compressors is currently 100 psig.

If Instrument Air pressure at the outlet of the compressors were to drop to 80 psig, what would be the status of the Instrument Air Compressors?

A. 2C-27A running loaded, 2C-27B running loaded

B. 2C-27A running loaded, 2C-27B running unloaded

C. 2C-27A running loaded, 2C-27B in standby

D. 2C-27A running unloaded, 2C-27B in standby

Answer:

A. 2C-27A running loaded, 2C-27B running loaded

Notes:

The LEAD compressor will load at 95 psig decreasing IA pressure and the standby LAG IA compressor will start prior to reaching its loading pressure of 85 psig. At 80 psig, both IA compressors should be fully loaded.

References:

ANO-2-LP-AO-IA, Revision 12, Objective 6
STM 2-48, Instrument Air, Revision 3, Sections 2.7 and 2.8.2
OP 2104.024, Instrument Air System Operation, Revision 030-04-0, Steps 6.1, 17.7, and 7.2.9 through 7.2.12 along with the note above step 7.2.9.

Historical Comments:

1/10/2002 Switch answer A and Distracter B; changed distracter D to A running unloaded, b in standby based on NRC feedback. BNC

QID:	0394	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	3.5	**Type:**	Containment Integrity				
System Title:	Containment System	**System Number**	103	**K/A:**	K4.06						
RO Tier:	2	**RO Group:**	3	**RO Imp:**	3.1	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	3.7

Description: Knowledge of containment system design feature(s) and/or interlock(s) which provide for the containment isolation system.

Question:

Given the following plant conditions:

* The plant has been tripped due to low SG pressure.
* SG #1 pressure is 850 psia and slowly recovering.
* SG #2 pressure is 725 psia and slowly dropping.
* Containment pressure is 19.3 psia and slowly rising.

At this point, the flowpath for RCP controlled bleed-off would be directed to:

A. The Volume Control Tank

B. The Reactor Drain Tank

C. The RCS Quench Tank

D. The Containment Sump

Answer:

C. The RCS Quench Tank

Notes:

The Containment Isolations for Controlled Bleed-off will close on a CIAS due to Containment Pressure above the setpoint of 18.3 psia. This will cause the controlled bleed-off pressure to rise to the controlled bleed-off relief setpoint of 2PSV-4836 set at 150 psig. This will direct controlled bleed-off to the RCS Quench Tank. The VCT is the normal path for the bleed-off flow path. The Reactor Drain Tank has the majority of CVCS and ECCS relief drain paths. The containment Sump collects all other drains inside containment.

References:

ANO-2-LP-RO-CVCS, Revision 8, Objectives 3 and 4
ANO-2-LP-RO-RCS, Revision 12, Objective 25
STM 2-04, CVCS, Revision 12, Section 2.1.21.3
STM 2-70, ESF System, Revision 7, ESFAS Actuation Description Table

Historical Comments:

QID:	0396	**Rev:**	000	**Rev Date:**	12/06/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct of Operations	**System Number**		**K/A:**	2.1.22						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	2.1	**SRO Imp:**	3.3
Description:	Ability to determine Mode of Operation.										

Question:

In accordance with OP 2102.016, Reactor Startup, when will the Shift Manager log into the Station Log that the Unit has entered into Mode 2

Entry in Mode 2 will be when:

A. All Shutdown Bank CEAs have been fully withdrawn.

B. Commencing withdrawal of Regulating Group 1 CEAs.

C. Regulating Group 2 CEAs have been fully withdrawn.

D. When the CBOR calls the reactor critical.

Answer:

C. Regulating Group 2 CEAs have been fully withdrawn

Notes:

Per the Reactor Startup procedure, OP 2102.016, step 8.5, Mode 2 is entered when the Regulating Group 2 CEAs are fully withdrawn.

References:

ANO-2-LP-RO-TS, Revision 7, Objective 1
OP 2102.016, Reactor Startup, Revision 007-02-0, Step 8.5
Technical Specifications Table 1.1

Historical Comments:

QID:	0397	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.2 & 43.5	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct of Operations	**System Number**		**K/A:**	2.1.12						
RO Tier:	3	**RO Group:**	2.1	**RO Imp:**	2.9	**SRO Tier:**	3	**SRO Group:**	1	**SRO Imp:**	4.0
Description:	Ability to apply technical specifications for a system.										

Question:

Given the following plant conditions:

* The plant is at 88% power and stable.
* COLSS is declared inoperable.
* No other equipment is out of service.
* The CRS has entered 2203.043, Loss of COLSS.
* The following data is collected from the CPC channels within the next 15 minutes.

	DNBR	LPD	ASI
CPC A	2.22	12.21	-0.051
CPC B	2.26	12.34	-0.057
CPC C	2.18	12.16	-0.062
CPC D	2.24	12.41	-0.057

The correct action to take based on this data would be to: (Reference Material Provided)

A. continue taking and monitoring data every 15 minutes, all LCOs are currently satisfied.

B. restore DNBR to within the COLR Limit within 2 hours or reduce power to < 20% in 6 Hours.

C. restore LPD to within the COLR Limit within 2 hours or reduce power to < 20% in 6 Hours.

D. immediately commence a power reduction due to DNBR/LPD exceeding calculated limits.

Answer:

B. restore DNBR to within the COLR Limit within 2 hours or reduce power to < 20% in 6 Hours.

Notes:

For closed Reference exam, a copy of T.S. 3.2.1, 3.2.4 and 3.2.7 should be provided along with COLR Figure 4 and applicable Limit steps.

Based on the given parameters and COLR Figure 4, DNBR is outside the acceptable range for 80% power so T.S. 3.2.4 action b applies. LPD is within the LCO limit of the COLR of <13.5. The limits calculated in the AOP for DNBR and LPD are the average of the 4 readings taken above plus or minus a margin so at this point the calculated limits are not exceeded and an immediate power reduction is not required.

References:

ANO-2-LP-RO-TS, Revision 7, Objective 4
OP 2203.043, Loss of COLSS, Revision 1, Steps 5 and 7 and Attachment A.
Technical Specifications 3.2.1, 3.2.4, and 3.2.7
Unit 2 COLR, Revision 0, Steps 6, 8, and 9, and Figure 4

Historical Comments:

QID:	0398	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.2	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct of Operations	**System Number**		**K/A:**	2.1.32						
RO Tier:	3	**RO Group:**	2.1	**RO Imp:**	3.4	**SRO Tier:**	3	**SRO Group:**	1	**SRO Imp:**	3.8
Description:	Ability to explain and apply all system limits and precautions.										

Question:

The Reactor Startup procedure, 2102.016, requires that the RCS average temperature remain greater than or equal to 525°F when the Reactor is critical.

One of the reasons for this limitation is to ensure:

A. Moderator Temperature Coefficient is within its analyzed temperature range.

B. Shutdown margin is adequately maintained during power escalation.

C. CEA rod worths are within their analyzed Core Reload assumptions.

D. Core Operating Limits Supervisory System (COLSS) is calculating POL margins.

Answer:

A. Moderator Temperature Coefficient is within its analyzed temperature range.

Notes:

Answers A, is the only reason above listed in the basis for TS 3.1.1.5 for maintaining the RCS above 525°F. This is where the limitation of OP 2102.016, Reactor Startup, Step 5.14 comes from. Shutdown margin is maintained by keeping CEAs above the Transient Insertion limits given in the COLR. CEA rod worth assumptions are based on normal operating temperature of 545°F. COLSS start calculating POLs at 15 % power and above.

References:

ANO-2-LP-RO-OPROC, Revision 8, Objective 3
OP 2102.016, Reactor Startup, Revision 007-02-0, Step 5.14
Technical Specification 3.1.1.5 LCO and Basis

Historical Comments:

QID:	0399	**Rev:**	000	**Rev Date:**	08/15/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.6	**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.27						
RO Tier:	3	**RO Group:**	2.2	**RO Imp:**	2.6	**SRO Tier:**	3	**SRO Group:**	2	**SRO Imp:**	3.5
Description:	Knowledge of the refueling process										

Question:

Given the following plant conditions:

* Plant is in Mode 6.
* Refueling shuffle is in progress from the Reactor to the Spent Fuel Pool (SFP).
* Refueling Cavity water level is 401 feet 6 inches.
* A Non Licensed Operator (NLO) informs the control room that both doors on the Containment access personnel airlock are wide open.

The correct response to this report would be to:

A. Immediately suspend all movement of irradiated fuel inside containment and at the SFP.

B. Immediately suspend core alterations and irradiated fuel movement inside Containment.

C. Have the NLO close one airlock door and inform security of the Containment Breech.

D. Have the NLO verify that either one of the airlock doors is capable of being closed.

Answer:

D. Have the NLO verify that either one of the airlock doors is capable of being closed.

Notes:

TS Amendment 230 to TS 3.9.4 allows the personnel airlock doors to remain open during core alterations as long as a minimum of one of the two doors is capable of being closed. This is a new amendment that no longer requires one of the airlock doors to be closed during core alteration. Since the LCO is met with either airlock door capable of being closed, then the actions to suspend core alterations or irradiated fuel movement do not apply.

References:

ANO-2-LP-RO-TS, Revision 7, Objective 4
OP 2502.001, Refueling Shuffle, Revision 029-04-0, Step 6.31
Technical Specification 3.9.4 - Amendment 230
STM 2-13, Containment Building, Revision 5, Section 7.13

Historical Comments:

QID:	0400	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**		**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.2						
RO Tier:	3	**RO Group:**	2.2	**RO Imp:**	4.0	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	3.5
Description:	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.										

Question:

Given the following plant conditions:

* The plant is being shutdown for a refueling outage from 100% power.
* All Control Element Assemblies (CEAs) are at the upper electrical limit.
* CRS directs CBOR to commence Axial Shape Index (ASI) control using Group P CEAs.

The correct control manipulations to control ASI using Group P CEAs would be:

A. Mode Select to "MG", Group Select to "P", P Group Select to "P", Joystick to "Insert".

B. Mode Select to "MS", Group Select to "P", P Group Select to "P", Joystick to "Insert".

C. Mode Select to "MG", Group Select to "P", P Group Select to "P2", Joystick to "Insert".

D. Mode Select to "MS", Group Select to "P", P Group Select to "P1", Joystick to "Insert".

Answer:

A. Mode Select to "MG", Group Select to "P", P Group Select to "P", Joystick to "Insert".

Notes:

Only Group 1 through 5 CEAs can be operated in Manual Sequential, MS, mode. Group P CEAs require the Manual Group (MG) Mode of operation and also requires the P Group Select switch to be in the "P" position so that both subgroups in group P move together. The P1 and P2 position were part of the original CE design when Part length CEAs were installed in the core. The P Group Select switch is now operational only when in the P position.

References:

ANO-2-LP-RO-CEDM, Revision 8, Objective 6
STM 2-02, Control Element Drive Mechanism Control System, Revision 7, Sections 4.2.1.2, 4.2.1.4, and 4.2.3.

Historical Comments:

QID:	0401	**Rev:**	000	**Rev Date:**	11/29/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.6	**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.33						
RO Tier:	3	**RO Group:**	2.2	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of control rod programming.										

Question:

Given the following:

* A Reactor Startup is in progress.
* Shutdown Banks A and B have been cocked.

During withdrawal of the Regulating Group Control Element Assemblies (CEAs) Groups 1 through 5 in Manual Sequential (MS) which ONE of the following systems controls the CEA group sequencing.

A. Control Element Assembly Calculators (CEACs) using CEA reed switch positioning.

B. Plant Monitoring System (PMS) using CEA pulse counting group positioning.

C. Control Element Drive Mechanism Control System Cabinets using CEA reed switches.

D. Plant Protection System (PPS) Computers using CEA pulse counting group positioning.

Answer:

B. Plant Monitoring System (PMS) using CEA pulse counting group positioning.

Notes:

The CEAC computers and PPS computers are for monitoring CEA position only and providing input to the plant safety limits calculation. They provide no control function. The CEDMCS cabinets provide the interface to the mag jack coils that provide the actual moving force to step in or out CEAs but do not control CEA sequencing permissives and group overlap. All the sequencing controls and group overlap comes from the PMS.

References:

ANO-2-LP-RO-CEDM, Revision 8, Objective 7
STM 2-02, Control Element Drive Mechanism Control System, Revision 8, Sections 4.2.1.4 and 4.2.1.6.

Historical Comments:

QID:	0405	**Rev:**	001	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**		**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.31						
RO Tier:	3	**RO Group:**	2.4	**RO Imp:**	3.3	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of annunciator alarms and indications, and use of the response instructions.										

Question:

Given the following:

* A Plant transient has occurred causing several annunciators to come in.

In accordance with the EOP/AOP Users Guide, 1015.021, the correct order to address and prioritize the annunciators based on their color coding would be:

A. Green, White, and Red

B. Red, Green, and White

C. Red, White and Green

D. Green, Red, and White

Answer:

B. Red, Green, and White

Notes:

ANO has three colors to code different annunciators based on their significance. Red (High Awareness) would be the highest priority to due the safety significance of the alarm. Green (Medium Awareness) would be next and White (General Awareness) would be last.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 1
OP 1015.021, ANO-2 EOP/AOP User Guide, Revision 004-02-0, Step 7.0 Annunciator Usage

Historical Comments:

1/10/2002. Removed the color yellow from distracter A. Added "In accordance with the EOP/AOP Users Guide, 1015.021," to the stem based on NRC feedback. 1/17/2002 - Removed the color yellow from all distracters and made the distracters a combination of red white and green to be more credible based on NRC feedback.

QID:	0406	**Rev:**	000	**Rev Date:**	10/08/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**		**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.19						
RO Tier:	3	**RO Group:**	2.4	**RO Imp:**	2.8	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of EOP layout, symbols, and icons										

Question:

Floating Steps within an EOP Optimal Recovery procedure can be identified by:

A. A black circle prior to the instructional step number.

B. A black asterisk prior to the instructional step number.

C. A black square prior to the instructional step number.

D. A black triangle prior to the instructional step number.

Answer:

C. A black rectangle prior to the instructional step number.

Notes:

FLOATING steps are procedural steps which require performance any time the specified condition exists and are prefaced by a solid black rectangle. A black circle would be a bulleted step in the preferential order of conductance. Procedural actions marked with an asterisk are CONTINUOUS ACTION steps. Included are steps which must be performed continuously after they are presented while the invoking EOP is in effect. Black triangles are not used to identify EOP steps or actions.

References:

ANO-2-LP-RO-ESPTA, Revision 5, Objective 5
OP 1015.021, Revision 004-02-0, Steps 4.8 and 4.18

Historical Comments:

QID:	0407	**Rev:**	000	**Rev Date:**	10/08/2001	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	Modified	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**		**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.15						
RO Tier:	3	**RO Group:**	2.4	**RO Imp:**	3.0	**SRO Tier:**	3	**SRO Group:**	4	**SRO Imp:**	3.5
Description:	Knowledge of communications procedures associated with EOP Implementation.										

Question:

Which ONE of the following defines the EOP verb VERIFY in the EOP/AOP Users Guide?

A. Observe that an expected condition exists, but does not permit action to make the condition occur.

B. Evaluate the status of a parameter to establish whether or not an action should be performed immediately.

C. Check the status of a process parameter within a given band repeatedly, at an unspecified interval.

D. Observe that an expected condition exists and, if it does not then take action to establish the condition.

Answer:

D. Observe that an expected condition exists and, if it does not then take action to establish the condition.

Notes:

Per the definition section, Attachment B of the Unit 2 EOP/AOP Users Guide, direction to verify a component allows the operator to take an action to align a component with the given direction if it is not already aligned. This make D the only correct answer.

References:

ANO-2-LP-RO-ESPTA, Revision 5, Objective 3
OP 1015.021, EOP/AOP Users Guide, Revision 004-02-0, Attachment B, Definition of Verify.

Historical Comments:

From the INPO Exam database QID 17586

QID:	0409	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Inoperable/Stuck Control Rod	**System Number**	005	**K/A:**	AA2.03						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.4
Description:	Ability to determine and interpret required actions if more than one rod is stuck or inoperable as they apply to the Inoperable/Stuck Control Rod.										

Question:

Given the following:

* A plant down power to <90% is being conducted to repair a Circulating Waterbox tube leak.
* ASI control is implemented by the CBOR using Group P CEAs.
* The CBOR reports that CEA #22 has slipped to 105 inches withdrawn.
* The CBOR also reports that CEA #28 Rod Bottom Light is lit.
* All other CEAs in Group P are at 130 inches withdrawn.

Which ONE of the following actions should be taken based on these conditions?

A. Manually trip the Reactor and go to 2202.001, Standard Post Trip Actions.

B. Stabilize the plant by adjusting turbine load to match T-REF within 2°F of T-AVE.

C. Continue the power reduction to 60% power and realign the CEAs.

D. Continue the power reduction to be in HOT STANDBY conditions in 6 hours.

Answer:

A. Manually trip the Reactor and go to 2202.001, Standard Post Trip Actions.

Notes:

Distracters A, B and D are actions required in the CEA Malfunction AOP or Technical Specifications based on different misalignment scenarios. However only answer C is the correct action to take if two or more CEAs are misaligned by more than 19 inches.

References:

ANO-2-LP-SRO-AOP, Revision 7, Objective 6
STM 2-02, CEDMCS, Revision 8, Section 1.2
OP 2203.003, CEA Malfunction, Revision 014-02-0, Continuous Action Step #4.

Historical Comments:

QID:	0412	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	Excessive Steam Demand	**System Number**	E05	**K/A:**	EA2.2						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.2
Description:	Ability to determine and interpret the adherence to procedures and operation within the limitations in the facility's license and amendments.										

Question:

Given the following:

* The plant has tripped from 100% Power.
* RCS pressure is 1600 psia and lowering.
* RCS T-cold is 505°F and lowering.
* Pressurizer Level is 10% and lowering.
* Containment pressure is 14.5 psia and stable.
* Containment temperature is 110°F and stable.
* No radiation alarms are present inside containment or on the Main Steam lines.
* A Steam Generator pressure is 610 psia and lowering.
* B Steam Generator pressure is 610 psia and lowering.
* A Steam Generator level is 20% NR and lowering.
* B Steam Generator level is 20% NR and lowering.
* No Main Steam Safeties have lifted.
* No other abnormal conditions exist and all components have actuated as designed.
* All systems function as designed.

Which ONE of the following actions should be taken to stabilize plant pressure and temperature?

A. Manually initiate a MSIS and verify the Main Steam Isolation Valves go closed.

B. Take manual control of the FWC system and minimize Main Feed to Steam Generators.

C. Close both Main Steam to the EFW Pump Terry Turbine, 2P7A, isolation valves.

D. Take manual control of the HPSI system and throttle the excess flow to the RCS.

Answer:

C. Close both Main Steam to the EFW Pump Terry Turbine, 2P7A, isolation valves.

Notes:

Answers A and B are both incorrect because a MSIS should have already occurred causing the MSIVs and Main Feed Isolations to close so an excessive steaming path downstream of the MSIVs or an excessive feeding to the SGs should not exist. Answer D is incorrect because even though a SIAS has been initiated, the RCS pressure is still above the shutoff head of a HPSI pump so excessive cooling flow from the HPSI pumps should not exist. Answer C is correct because the steam isolations to the Terry Turbine are upstream of the MSIVs, outside containment and they cross connect both Steam Generators.

References:

ANO-2-LP-RO-EESD, Revision 2, Objectives 09.3 and 09.4
OP 2202.005, Excess Steam Demand EOP, Revision 005-01-0, Floating Step 15
EOP 2202.005, Technical Guide, Revision 005-01-0, Step 15

Historical Comments:

QID:	0413	**Rev:**	000	**Rev Date:**	01/15/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 41.10	**10CFR55_43:**		**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Inadequate. Core Cooling	**System Number**	074	**K/A:**	EK3.11						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.4
Description:	Knowledge of the reasons for the guidance contained in EOP as they apply to Inadequate Core Cooling.										

Question:

Given the following:

* The plant has tripped due to a Loss of Offsite Power 1 hour ago.
* A bus lockout occurs on Electrical Bus 2A3 and cannot be reset.
* EFW Pump 2P7A trips on overspeed and cannot be reset.
* The Loss of Feed Water ORP, 2202.006 has been entered.
* RCS pressure is 2100 psia and rising.
* RCS T-cold is rising at an uncontrolled rate.
* Both Steam Generator levels are 95 inches and dropping.

The correct action to take based on these conditions would be to:

A. Establish Once Through Cooling with SI flow to remove RCS heat at this time.

B. Establish Once Through Cooling to remove RCS heat at < 70 inches in either SG.

C. Transition to the Functional Recovery Procedure to establish RCS heat removal.

D. Transition to Once Through Cooling for RCS heat removal after SGs are dry.

Answer:

A. Establish Once Through Cooling with SI flow to remove RCS heat at this time.

Notes:

By the Guidance found in the Loss of Feedwater EOP 2202.006, Once Through Cooling should be established when either SG is < 70 inches or RCS T-cold is rising in an uncontrolled manner. Once Through Cooling should be established before transitioning to the FRP.

This question was generated from a randomly selected K/A to be part of the SRO exam and not on the RO exam; however, this question is not one of the 25 10 CFR 55.43 category questions selected for this exam. Four additional questions were selected to be on the SRO exam that are not on the RO exam to in order to comply with the NUREG 1021 guidance to have a balance of K&A selections on the initial sample plan. One of these 4 happen to fall into the 10 CFR 43 category so there are actually 26 SRO only questions on the SRO exam that are in the 10 CFR 43 category.

References:

ANO-2-LP-RO-ELOSF, Revision 1, Objective 5
OP 2202.006, Loss of Feedwater EOP, Revision 005-01-0, Step 19
EOP 2202.006, Technical Guidance, Revision 005-01-0, Step 19

Historical Comments:

QID:	0415	**Rev:**	000	**Rev Date:**	10/30/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	ARM System Alarms	**System Number**	061	**K/A:**	AA2.06						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.1
Description:	Ability to determine and interpret required actions if alarm channel is out of service.										

Question:

Given the following:

* The plant is at 100% Power steady state.
* Technical Specification 3.4.6.2, RCS Leakage, requires a RCS Water Inventory Balance surveillance be completed once every 72 hours in Modes 1 through 4.
* Chemistry is not available to perform any samples at this time.

Which ONE of the following radiation monitors being inoperable would require the frequency of the RCS Water inventory balance surveillance to be increased to once every 24 hours?

A. Containment High Range Radiation Monitors 2RITS-8925-1 and 2RITS-8925-2.

B. Containment elevation 404 South West end of the Refueling Deck 2RITS-8912.

C. Containment Atmosphere Monitors (CAMS) 2RITS-8231-1 and 2RITS8271-2.

D. Containment Purge Exhaust, 2VEF-15 Discharge, Radiation Monitor 2RITS-8233.

Answer:

C. Containment Atmosphere Monitors (CAMS) 2RITS-8231-1 and 2RITS8271-2.

Notes:

Technical Specification 3.4.6.1 requires a containment atmosphere particulate and gaseous radioactivity monitor to be operable or the action requires grab samples or an RCS leak rate every 24 hours. Both Containment High Range monitors being OOS require alternate method of monitoring within 72 hours. Answer B requires an alternate method of monitoring if refueling operations is in progress. Answer D requires discontinuation of containment Purge which should not be in operation in Mode 1.

References:

ANO-2-LP-RO-RMON, Revision 10, Objective 18
Technical Specification 3.4.6.1, Amendment 231, Action A
Technical Specification 3.4.6.2, Amendment 184 surveillance 4.4.6.2.1

Historical Comments:

QID:	0416	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Instrument Air	**System Number**	065	**K/A:**	AA2.06						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.2
Description:	Ability to determine and interpret when to trip the reactor if instrument air pressure is decreasing as it applies to Loss of Instrument Air.										

Question:

Given the following:

* The plant is at 100% Power.
* Annunciator 2K12 A-8, INSTR AIR PRESS HI/LO, comes in.
* The CBOT reports that Instrument Air (IA) Header Pressure is 75 psig and dropping.

Which ONE of the following is the correct action to take if IA Header Pressure continues to drop?

A. At 35 psig, trip the Reactor and commence Standard Post Trip Actions.

B. At 40 psig, align Service Water to the Containment fan cooling units.

C. At 60 psig, open the IA cross-connect isolation valves from Unit 1.

D. At 65 psig, start the temporary IA Compressor and enter Loss of IA AOP.

Answer:

A. At 35 psig, trip the Reactor and commence Standard Post Trip Actions.

Notes:

The Loss of IA AOP entry conditions is 80 psig IA pressure and lowering so the AOP should have already been entered in answer A. Answer B is wrong because the cross-connect isolations should be opened at 80 psig and dropping and then closed at 60 psig and dropping to prevent a loss of IA on Unit 1. A loss of IA will cause an isolation of Main Chilled Water to the Containment Cooling Units so an option would be to line up Service Water to the units; however, this is not a step directed by the Loss of IA AOP thus answer C is wrong. The procedure directs tripping the unit at 35 psig IA header pressure.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 16
OP 2203.012L, Annunciator 2K12 Corrective Action, Revision 030-02-0, Window A-8, IA Press Hi/LO
OP 2203.021, Loss of IA AOP, Revision 008-01-0, Entry Conditions, Step 4, and Step 5.

Historical Comments:

QID:	0417	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	CE EPE/APE				
System Title:	Excess RCS Leakage	**System Number**	A16	**K/A:**	2.4.1						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	3	**SRO Imp:**	4.6
Description:	Knowledge of the EOP Entry conditions and immediate actions steps as related to Excess RCS Leakage.										

Question:

Given the following:

* The plant is at 100% power when an RCS leak develops.
* The Crew enters OP 2203.016, Excess RCS Leakage AOP.
* A RCS leak rate calculates 90 gpm RCS leakage and steady.
* Letdown is isolated.
* Pressurizer Level is 48%.
* The Swing Charging Pump 2P36C is powered from the red train.
* A 2A4 Electrical Bus lockout occurs.

Which ONE of the following describes the correct action to take at this point?

A. Commence a controlled plant shutdown and cooldown to Mode 5 conditions.

B. Stabilize plant power and restore the 2A4 Electrical Bus before continuing.

C. Trip the Reactor, commence SPTAs, and enter the LOCA Recovery Procedure.

D. Start a High Pressure Safety Injection Pump and shutdown to Hot Standby.

Answer:

C. Trip the Reactor, commence SPTAs, and enter the LOCA Recovery Procedure.

Notes:

With the loss of 2A4, only 2 charging pumps are available for makeup with a capacity of 88 gpm. The exit conditions for the RCS Excess leakage AOP is when the leak exceeds the charging pump capacity which is the case here. Step 4 of the AOP directs a Reactor trip and SPTAs when Pressurizer Level is > 10% deviated from setpoint (60% at 100% power) and still dropping with all available charging pumps running. A controlled plant shutdown is directed by the AOP only if within the capacity of the charging pumps. A RCS leak can be mitigated with one safety bus so the leak takes precedence over restoring the electrical safety bus. A HPSI pump cannot provide the additional makeup required at normal plant pressure during the shutdown.

References:

ANO-2-LP-RO-EAOP, Revision 5, Objective 11
OP 2203.016, Excess RCS Leakage AOP, Revision 009-01-0, Exit Conditions and Step 4.

Historical Comments:

QID:	0418	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	An	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	Reactor Trip/Recovery	**System Number**	E02	**K/A:**	EA2.1						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	3.7

Description: Ability to determine and interpret facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to the Reactor Trip/Recovery.

Question:

Given the following conditions:

* The Reactor has tripped.
* No Safety or Non-Safety 4160 VAC and 6900 VAC busses are energized.
* RCS Tcold is 555F and stable.
* Steam Generator levels are 15% and slowly trending down.
* Standard Post Trip Actions (SPTAs) are complete.
* The Crew has entered OP 2202.008, Station Blackout recovery procedure.
* Main Steam Isolation valves have been closed manually.
* EFW Pump 2P7A has tripped on overspeed and cannot be reset.

Which ONE of the following correctly describes the action the CRS should take?

A. Re-perform the Standard Post Trip Actions and re-diagnose the event.

B. Continue in the Station Blackout recovery procedure until <70 Inches SG Levels.

C. Exit the Blackout procedure and enter the Functional Recovery procedure.

D. Exit the Blackout procedure and enter the Loss of Feedwater procedure.

Answer:

C. Exit the Blackout procedure and enter the Functional Recovery procedure.

Notes:

With no feedwater available to feed the Steam Generators, the Station Blackout Procedure safety function for RCS Heat Removal is not being met. With two events and actions in the Blackout procedure not satisfying the respective safety functions, the EOP users guide states that the Functional Recovery Procedure (FRP) must be used (OP 1015.021 - Step 5.1.8 C.). Performing the SPTAs again will not help; however, Step 7E of OP 2202.008 sends the CRS directly back to diagnostics with out performing SPTAs and the diagnostics should send the crew to the FRP. The Crew cannot remain in the selected Optimum Recovery Procedure with out meeting all the safety functions. The Loss of Main Feedwater Procedure will not be effective without power.

References:

ANO-2-LP-RO-EFRP, Revision 2, Objective 1
OP 2202.008, Station Blackout, Revision 005-01-0, Step 7E and RCS Heat Removal Safety Function #6

OP 1015.021, Unit 2 EOP/AOP Users Guide, Revision 004-02-0, Step 5.1.8
OP 2202.010, Standard Attachments, Revision 006-00-0, Attachment 3, Diagnostic Actions

Historical Comments:

QID:	0419	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct Of Operations	**System Number**		**K/A:**	2.1.7						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	1	**SRO Imp:**	4.4
Description:	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.										

Question:

Given the following.

* Unit 2 is in Mode 5.
* 2B5 Bus Voltage is 482 Volts.
* 2B5 supply amps are 65 amps.
* 2B6 Bus Voltage is 478 Volts.
* 2B6 supply amps are 75 amps.
* Each bus (2B5 & 2B6) are supplied from their respective 4160V bus.
* All house loads are being supplied from #3 SU Transformer.
* It is desired to cross-connect 2B5 from 2B6 with both buses remaining energized.

Which ONE of the following action would be necessary to perform the cross-connect?

A. Reduce loads on 2B5/6 to ensure transformer loading is <130 amps.

B. Raise 2B6 bus voltage to 480 Volts prior to cross-connecting busses.

C. Reduce 2B5 bus voltage to 480 Volts prior to cross-connecting busses.

D. Raise loads on 2B5/6 to ensure transformer loading is > 150 amps.

Answer:

A. Remove loads on 2B5/6 to reduce transformer loading to <130 amps.

Notes:

The limitation on cross-tying the ESF 480 Volt busses is a total amp draw of 130 amps to prevent damage to either supply transformer. Voltages on the 480 Volt ESF busses are adequate if maintained greater than 438 volts. The transformer limit would be exceeded in answer D

References:

ANO-2-LP-RO-ED480, Revision 1, Objective 10
STM 2-32-3, 480 Volt Distribution System, Revision 3, Section 3.0
2107.002, ESF Electrical System Operations, Revision 015-02-0, Step 5.2 and Attachment "K" Step 4.0

Historical Comments:

QID:	0420	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.1	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct Of Operations	**System Number**		**K/A:**	2.1.10						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	1	**SRO Imp:**	3.9
Description:	Knowledge of conditions and limitations in the facility license.										

Question:

Given the following:

* The plant is at 100% Power.
* A Safety Limit has been exceeded.

The plant shall be placed in at least Hot Standby within ________ hour(s) and the Vice President of ANO Operations along with the Safety Review Committee (SRC) shall be notified of the violation within ________ hours.

A. 6; 24

B. 6; 30

C. 1; 24

D. 1; 30

Answer:

C. 1; 24

Notes:

Most Technical Specification action give 1 hour to correct the plant to within the LCO and then 6 hours to Hot Standby and 30 hours to cold shutdown. This makes answers A, B and C viable answers but they are incorrect. A violation of a Safety Limit requires the plant to be in at least Hot Standby in 1 hour and notification of the Vice President/Safety Review Committee within 24 hours.

References:

ANO-2-LP-SRO-TS, Revision 07, Objectives 2 and 10.
Technical Specifications, Administrative Controls, Section 6.7.1, Amendment 218

Historical Comments:

1/10/2002. Spelled out SRC (Safety Review Committee) based on NRC feedback

QID:	0421	**Rev:**	000	**Rev Date:**	12/08/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct Of Operations	**System Number**		**K/A:**	2.1.22						
RO Tier:	3	**RO Group:**	2.1	**RO Imp:**	2.8	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine Mode of Operation.										

Question:

Given the following:

* The plant has been shutdown for 3 days to conduct a refueling outage.
* The plant is in Mode 5.
* Shutdown Cooling is in service.
* RCS temperature is 135 degrees.
* Keff has been calculated to be 0.92.
* RCS Boron Concentration is 1500 ppm.
* The Refueling Team has just completed removing the Reactor Vessel Maintenance Structure

Mode 6 will be entered when:

A. RCS Temperature has been reduced below 125°F.

B. Reactor Vessel head bolts have been detensioned.

C. Keff has been reduced below 0.90.

D. RCS and Refueling Canal Boron concentration is > 2500 ppm.

Answer:

B. Reactor Vessel head bolts have been detensioned.

Notes:

Mode 6 is defined as Keff less than or equal to 0.95 with RCS temperature less than or equal to 140°F. In this question the RCS temperature and Keff already meet the Mode 6 requirements so that all that is left to enter Mode 6 is detensioning the Reactor Vessel Head. 2500 ppm boron concentration is the requirement to commence a fuel shuffle from the core to the Spent Fuel Pool.

References:

ANO-2-LP-SRO-TS, Revision 7, Objective 1
Technical Specifications, Definition of Mode 6, Amendment 60
OP 2504.007, Unit 2 Reactor Vessel Stud Detensioning and Storage, Revision 011-00-0, Steps 3.0 and 6.2.5.

Historical Comments:

QID:	0422	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	Modified	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.3	**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.9						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	2	**SRO Imp:**	3.3
Description:	Knowledge of the process for determining if the proposed change, test or experiment increases the probability of occurrence or consequence of an accident during the change, test, or experiment.										

Question:

As the responsible supervisor, you are performing an INTERIM approval for a permanent procedure change (PC) required to continue a HPSI pump quarterly surveillance conducted on the weekend. The 50.59 SCREENING for this PC indicates that a 50.59 EVALUATION must be completed.

Which of the following statements describes the correct action concerning the procedure change:

A. Approval can be granted as long as the OSRC, Onsite Safety Review Committee, reviews the 50.59 EVALUATION in fourteen days.

B. Do not approve it because a 50.59 EVALUATION is required.

C. Approval can be granted without completion of the 50.59 EVALUATION.

D. Do not approve it because a special OSRC, Onsite Safety Review Committee, must be called for approval.

Answer:

B. Do not approve it because a 50.59 EVALUATION is required.

Notes:

A SRO cannot approve an interim procedure change if the 50.59 screening requires a 50.59 evaluation since the change could affect a license bases document and therefore requires more scrutiny, additional reviews, prior to implementation. A standard procedure change process must be implemented.

References:

OP 1000.006, Procedure Control section 7.10, rev. 050-02-0
NMM LI-101, ATTACHMENT 9.1, 50.59 Review Form, rev. 1

Historical Comments:

ANO Exam Bank.
1/10/2002. Reworded distracters A and C based on suggested feedback from the NRC. BNC

Questions For 2002 SRO/RO Exam

17-Jan-02

QID:	0423	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.6	**Section:**	2.2	**Type:**	Generic K/A				
System Title:	Equipment Control	**System Number**		**K/A:**	2.2.32						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	2	**SRO Imp:**	3.3
Description:	Knowledge of the effects of alterations on core configuration.										

Question:

Given the following:

* Mode 6; core reload in progress.
* Due to fuel assembly bowing, the SRO in-charge of refueling requests that the fuel assembly be temporarily stored in one of the reactor building fuel storage racks and loaded into the core later in the refueling shuffle sequence.

Which of the following must be done prior to the change in the refueling shuffle sequence:

A. Change recorded on refueling shuffle procedure, 2502.001, Attachment 'B'.

B. Written approval from the Operations Manager and Reactor Engineering supervisor.

C. A complete core reload analysis performed and approved.

D. A refueling shuffle, 2502.001, procedure change must be submitted and approved.

Answer:

A. Change recorded on refueling shuffle procedure, 2502.001, Attachment 'B'.

Notes:

Procedure 2502.001, Refueling shuffle, allows for changes in the core off-load or on-load sequence without a procedure change, core reload analysis (so long as the specified fuel assembly is not replaced), nor written approval. The only requirement is that the change be documented on 2502.001 attachment B.

References:

OP 2502.001, refueling shuffle, limit and precaution 6.6. Rev. 029-04-0
OP 2502.001, refueling shuffle, attachment 'B'. Rev. 029-04-0

Historical Comments:

QID:	0425	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K/A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.6						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	3	**SRO Imp:**	3.1
Description:	Knowledge of the requirements for reviewing and approving release permits.										

Question:

An NRC commitment exists that requires a 30 day isolation of a gaseous decay tank (2T18) prior to release. Which of the following best describes the reason for this requirement:

A. This requirement allows short lived gaseous activity present in the tank to decay.

B. This requirement ensures that all the radioactive gases have sufficient time to diffuse throughout the tank.

C. This requirement prevents purging the RCS to an isolated tank within the last 30 days.

D. This requirement ensures that an explosive mixture is not present in the tank during discharge.

Answer:

A. This requirement allows short lived gaseous activity present in the tank to decay.

Notes:

Distracter 'B' is incorrect since any gas present in the tank will readily diffuse throughout the tank.
Distracter 'C' is incorrect since this requirement is applicable for any release, since short lived decay products are present in the VCT.
Distracter 'D' is incorrect since oxygen and hydrogen readily diffuse throughout the tank and cannot collect in any 'pocket' within the tank that would necessitate a long wait period for diffusion.

References:

OP 2104.022, Gaseous Radwaste System Supplement 1, rev. 032-04-0
Commitment P-2244, 30-day hold up time on gaseous radioactive release.

Historical Comments:

QID:	0429	**Rev:**	000	**Rev Date:**	12/07/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	An	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.29						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	4	**SRO Imp:**	4.0
Description:	Knowledge of the emergency plan.										

Question:

Given the following plant conditions:

* Plant has tripped with LOCA in progress.
* Off site radiological release is in progress.
* Site Area Emergency was JUST declared.
* Radiation levels in the Unit 2 turbine building are 5.0 mRem/hr.
* Radiation levels in the Exclusion area are 0.1 mRem/hr.
* Unit 2 SM has E-Plan command and Control.

Which of the following would be the correct response for the given conditions:

A. Perform Plant evacuation (Protected Area).

B. Perform localized evacuation of Turbine Building.

C. Perform Exclusion Area evacuation.

D. Shelter all personnel, no evacuation is required.

Answer:

A. Perform Plant evacuation.

Notes:

'A' is correct since at the onset of a SAE declaration a plant evacuation is required.
'B' is incorrect although it is prudent, a plant evacuation will encompass a turbine building evacuation.
'C' is incorrect since criteria for exclusion area evacuation is not met.
'D' is incorrect since procedure requires a plant evacuation at SAE.

References:

1903.030, Evacuation, section 6.0; rev. 24-02-0
1903.011P, Emergency response and notifications, SAE emergency direction and control SM checklist, rev 26-00-0

Historical Comments:

QID:	0430	**Rev:**	000	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	An	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.33						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	4	**SRO Imp:**	2.8
Description:	Knowledge of the process used to track inoperable alarms.										

Question:

Given the following plant conditions:

* 100% power; all systems in normal configuration.
* Annunciator 2K12 J-1, Letdown Radmonitor Flow LO, alarms every 15 seconds.
* WCO verified local flow indication to be 0.7 gpm.
* Alarm setpoint verified to be > 0.5 gpm.

Which of the following is the preferred method to handle the above nuisance alarm until maintenance can be performed:

A. Isolate flow to the radiation monitor.

B. Have WCO raise flow until alarm clears, up to flow limit.

C. Pull the card for the annunciator.

D. Station additional operator in control room to acknowledge alarm.

Answer:

B. Have WCO raise flow until alarm clears, up to flow limit.

Notes:

Distracter 'A' is incorrect although it will stop the nuisance alarm it will prevent monitoring RCS failed fuel.
Answer 'B' is correct as the nuisance alarm is cleared and monitoring function is maintained.
Distracter 'C' is incorrect because it is less preferred than raising the process flow since it eliminates the monitoring of flow through the rad monitor by alarm function.
Distracter 'D' is incorrect since this will add congestion to the control room and the horn and light will still be acknowledged.

References:

OP 1015.028, operations annunciator control, rev. 005-03-0
ACA 2203.012L, ACA for 2K12 J-1, rev 030-02-0
OP 2104.002, CVCS procedure section 15.0, rev 039-00-0

Historical Comments:

1/10/2002. Corrected note errors based on NRC feedback. BNC

QID:	0431	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	4	**Taxonomy:**	K	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Vital AC Instrument Bus	**System Number**	057	**K/A:**	AA2.05						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	3.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine and interpret S/G pressure and level meters as they apply to the Loss of a Vital AC Instrument Bus.										

Question:

Given the following plant conditions:

* 100% power and normal system alignment.
* A loss of 2RS-2 occurs.

Which of the following SG level transmitters on each Steam Generator would still be available:

A. THREE channels of narrow range AND ONE wide range.

B. THREE channels of narrow range AND TWO wide range.

C. FOUR channels of narrow range AND ONE wide range.

D. FOUR channels of narrow range AND TWO wide range.

Answer:

A. THREE channels of narrow range AND ONE wide range.

Notes:

2RS-2 supplies power to channel 2 narrow range and wide range level transmitters. There are four narrow range safety channels and two wide range level instruments. Therefore a loss of 2RS 2 will result in three narrow range safety channels and one wide range level instruments remaining.

References:

OP 2107.008, Inverter and 120 VAC bus outage procedure. Attachment B, Rev. 002-02-0

Historical Comments:

QID:	0432	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Nuclear Service Water	**System Number**	062	**K/A:**	AA2.06						
RO Tier:	1	**RO Group:**	1	**RO Imp:**	2.8	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine and interpret the length of time after the loss of Service Water System flow to a component before the component may be damaged.										

Question:

Given the following plant conditions:

* 100% power, normal system alignment.
* Loop 1 SW is supplying CCW and is not cross-connected.
* Loop 2 SW is supplying ACW.
* #1 EDG is paralleled with off site power at 2850 KW.
* All systems are operating within normal temperature and pressure bands.

If 'A' Service water pump tripped and no operator action were to occur, which of the following would occur first:

A. #1 EDG scavenging air temp would exceed 180°F.

B. Letdown DI would be bypassed automatically.

C. SWC turbine runback.

D. RCP motor/bearing high temperature alarms.

Answer:

A. #1 EDG scavenging air temp would exceed 180°F.

Notes:

'A' is correct since the EDG is operating at full load and the EOP requires securing the EDG within 3 minutes if SW is lost.
Distracter 'B' is incorrect since it is cooled by CCW and the CCW system would have to heat up and then letdown would heat up which would all take longer than the EDG scavenging air temp.
Distracter 'C' is incorrect since ACW cools SWC heat exchanger and loop 2 is supplying ACW which is unaffected.
Distracter ' D' is incorrect since CCW cools these components and would take longer than the EDG scavenging air temperature to heat up to alarm setpoint and the AOP estimates 10 minutes on loss of CCW cooling flow before a RCP must be tripped..

References:

2203.025, RCP emergencies, Technical Guidelines, Rev. 8
2203.025, RCP emergencies AOP, rev. 8
2202.0003, LOCA EOP rev 005-01-0

Historical Comments:

QID:	0433	**Rev:**	000	**Rev Date:**	10/03/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	C	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Residual Heat Removal System	**System Number**	025	**K/A:**	AA2.04						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.3	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine and interpret the location and isolability of leaks as they apply to the Loss of Residual Heat Removal System.										

Question:

Given the following plant conditions:

* 'A' LPSI pump through 'A' SDC HX in service.
* RCS level is 24" above the bottom of the RCS hot leg and dropping rapidly.
* RCS temperature is 137 °F.

Which of the following should be performed?

A. Close SDC suction valve 2CV-5086-2

B. Stop 'A' LPSI pump

C. Close the LPSI flow control valve 2CV 5091

D. Start 'B' LPSI pump

Answer:

B. Stop 'A' LPSI pump

Notes:

Distracter 'A' is incorrect since the valve is deenergized open when in reduced inventory (2CV 5038-1 is available as an isolation to the SDC system). Isolation of SDC suction is desired, but only after the SDC pump is secured.
Distracter 'B' is correct since the possibility of cavitation is a concern.
Distracter 'C' is incorrect since isolation of the flow controller will not help mitigate the loss of inventory.
Distracter 'D' is incorrect since starting another SDC pump will only increase the possibility of cavitating the LPSI pumps and possibly gas binding the SDC system.

References:

AOP 2203.029, loss of SDC step 4, technical guidelines rev 10-03-0
AOP 2203.029, loss of SDC step 4, rev 10-03-0

Historical Comments:

QID:	0434	**Rev:**	000	**Rev Date:**	10/02/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.1	**Type:**	Generic EPE				
System Title:	Steam Generator Tube Rupture	**System Number**	038	**K/A:**	EA2.08						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.8	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine viable alternatives for placing plant in a safe condition when condenser is not available when SGTR present.										

Question:

Given the following plant conditions:

* Reactor tripped with a SGTR in progress.
* SIAS, CCAS have been actuated.
* The ruptured SG has been isolated in accordance with 2202.010 Att. 10, SG Isolation.
* HPSI has been overridden and SIAS has been reset.
* RCS pressure is 1200 psia.
* Ruptured SG pressure is 900 psia.
* Ruptured SG level is 47%.
* RCS cooldown in progress with 'A' and 'C' RCP's running.
* Condenser pressure is atmospheric.

The best method to cooldown and depressurize the ruptured SG given the above conditions is which of the following (assume all procedural steps to accomplish task have been completed unless precluded by above conditions):

A. Open the SG blowdown valves and EFW valves and feed and bleed SG.

B. Steam the SG using the upstream ADV's and feed with EFW.

C. Let the SG cool by ambient heat losses.

D. Lower RCS pressure to allow SG to drain to the RCS and expose SG tubes.

Answer:

D. Lower RCS pressure to allow SG to drain to the RCS and expose SG tubes.

Notes:

'A' is incorrect since condenser is not available, RCP's are running and it would use too much inventory.
'B' is incorrect since this would be an offsite radioactive release and the ruptured SG is well below the pressure that might lift an MSSV and ruptured SG level is not in danger or overfill concerns.
'C; is incorrect since this would take an inordinate amount of time resulting in unnecessary RCS inventory continuing to leak into the ruptured SG.
'D' is the preferred method described in the EOP.

References:

2202.004, SGTR, rev. 5 technical guidelines
2202.004, SGTR, rev. 5 step 47

Historical Comments:

QID:	0435	**Rev:**	000	**Rev Date:**	10/25/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Blanchard		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Accidental Liquid Radwaste Release	**System Number**	059	**K/A:**	AA2.04						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.2	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to determine and interpret the valve lineup for a release of radioactive fluid as they apply to the Accidental Liquid Radwaste Release.										

Question:

Given the following plant conditions:

* Boric Acid Condensate Tank, 2T69A, has been placed on recirc for a sample (30 minutes ago).
* Annunciator 2K11-C10, PROCESS LIQUID RADIATION Hi/LO actuates.
* 2RITS-2330, Liquid Rad Waste Disch to Flume, count rate is increasing above alarm setpoint.
* Reactor Drain Tank (RDT) draining evolution to Hold Up Tank 2T-12A is in progress.
* All other systems are in normal valve configuration.

Which of the following should be done immediately to mitigate the radioactive liquid release?

A. Contact Unit 1 and have them start another circulating water pump.

B. Close Liquid Release Isolations 2CV-2330A/B by taking 2RITS-2330 selector switch to the check source position.

C. Secure draining operations from the RDT to 2T-12A.

D. Contact WCO and have him check Boric Acid Condensate Pump, 2P-47A, recirc valve fully open or secure 2P-47A.

Answer:

D. Contact WCO and have him check Boric Acid Condensate Pump, 2P-47A, recirc valve fully open or secure 2P-47A.

Notes:

Distracter 'A' is incorrect since it will not stop the accidental release, however it will minimize the release rate due to raising the dilution factor.
Distracter 'B' is incorrect since the radiation monitor is already in alarm and the valve should have already closed.
Distracter 'C' is incorrect since the flow path from the RDT to the 2T12 tanks cannot be aligned to the liquid radwaste header.
'D' is the correct answer. Given the above conditions, the only source of radioactive liquid is from the 2T69A recirc lineup with leakby past the discharge valve

References:

M-2214 sheet 1, rev 64
M-2214 sheet 2, rev 64
2203.012K, ACA for 2K11 page 92 rev 29-04-0

Historical Comments:

QID:	0436	**Rev:**	000	**Rev Date:**	01/15/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5 & 41.10	**10CFR55_43:**		**Section:**	4.2	**Type:**	Generic APE				
System Title:	Pressurizer Pressure Control System Malfunct	**System Number**	027	**K/A:**	AK3.03						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.1
Description:	Knowledge of the reasons for actions contained in AOP for Pressurizer Pressure Control Malfunctions.										

Question:

Given the following:

* Plant is at 100% Power.
* A Pressurizer(PZR) Pressure malfunction has occurred.
* Procedure 2203.028, PZR Systems Malfunction, has been entered

If both Pressurizer Pressure Control Channels have failed, the procedure directs the following actions:

* Place the Steam Dump Bypass Control System, SDBCS, Master controller in AUTO LOCAL and adjust setpoint to 1000 psia.
* Verify a maximum of one 11.5% SDBCS Bypass or Downstream Atmospheric Dump Valves, ADV, Permissive switch in MANUAL.
* Verify all other SDBCS Bypass and ADV Permissive switches in OFF

The reason for the actions above are to:

A. Prevent spurious Quick Open signals to ALL SDBCS Bypass and ADV valves due to the failed PZR pressure bias input to the SDBCS Main calculator setpoint.

B. Limit the plant effects of any spurious open signals to the SDBCS Bypass and ADV valves due to failed PZR pressure bias input to the SDBCS Main and Permissive calculator setpoint.

C. Prevent spurious Modulation Open signals to ALL SDBCS Bypass and ADV valves due to failed PZR pressure bias input to the SDBCS Permissive calculator setpoint.

D. Limit the plant effects of any spurious open signals to the SDBCS Bypass valves due to failed PZR pressure bias input to the SDBCS Main calculator setpoint only.

Answer:

B. Limit the plant effects of any spurious open signals to the SDBCS Bypass and ADV valves due to the failed PZR pressure bias input to the SDBCS Main and Permissive calculator setpoint.

Notes:

Pressurizer Pressure Control Channel 1 inputs a bias to the SDBCS Main calculator while Pressurizer Pressure Control Channel 2 inputs to the SDBCS Permissive calculator. Since both Pressurizer Pressure Control channels have failed, all the ADVs and Bypass dump valves have the potential to open spuriously either by quick opening or modulating due to incorrectly calculated setpoints on the Main and Permissive calculators. By taking these actions the effect to the plant will be limited to one SDBCS or ADV valve with a 11.5% steam flow capacity due to the permissive for the selected valve in manual with a setpoint from the master at 1000 psia setpoint. This makes answer C correct and the others wrong.

This question was generated from a randomly selected K/A to be part of the SRO exam and not on the RO exam; however, this question is not one of the 25 10 CFR 55.43 category questions selected for this exam. Four additional questions were selected to be on the SRO exam that are not on the RO exam to in order to comply with the NUREG 1021 guidance to have a balance of K&A selections on the initial sample plan. One of these 4 happen to fall into the 10 CFR 43 category so there are actually 26 SRO only questions on the SRO exam that are in the 10 CFR 43 category.

References:

AN0-2-LP-RO-EAOP, Revision 5, Objective 21
OP 2203.028, PZR System Malfunction, Revision 5, Step 5
AOPP 2203.028, PZR System Malfunction Technical Guide, Revision 5, Step 5
STM 2-23, SDBCS STM, Revision 6, Section 6.2

Historical Comments:

1/10/2002. Reworded distracter D due to NRC feedback because of D was not credible answer and cues answer. BNC

QID:	0438	**Rev:**	0	**Rev Date:**	12/06/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	Natural Circulation Operations	**System Number**	A13	**K/A:**	AA2.2						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.8
Description:	Ability to determine and interpret adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.										

Question:

Given the following:

* A Loss of offsite power has occurred.
* All systems respond as designed.
* The plant transient caused a Steam Generator Tube Rupture(SGTR) on the 'A' SG.
* SGTR Optimum Recovery Procedure 2202.004, has been entered.
* The 'A' Steam Generator has been isolated.
* A natural circulation cooldown to Shutdown Cooling entry conditions is in progress.
* Offsite Power is now restored.
* The decision has been made to restart forced circulation
* All Reactor Coolant Pump (RCP) restart criteria has been met

The correct sequence for restarting RCPs and the reason for this sequence is:

A. Start 1 RCP in the ruptured loop, wait 5 minutes, then start 1 RCP in the intact Loop 2, to prevent thermal shock to SG 'A'

B. Start 1 RCP in the intact loop, wait 5 minutes, then start 1 RCP in the ruptured loop, to prevent rapid boron dilution in the reactor.

C. Start 2 RCPs in the ruptured loop, wait 30 minutes , then start 1 RCP in the intact loop, to prevent rapid boron dilution in the reactor.

D. Start 2 RCPs in the intact loop, wait 30 minutes, then start 1 RCP in the ruptured loop, to prevent thermal shock to SG 'A'

Answer:

B. Start 1 RCP in the intact loop, wait 5 minutes, then start 1 RCP in the ruptured loop, to prevent rapid boron dilution in the reactor.

Notes:

A RCP in the intact (least affected) loop should be started first and flow should be allowed to stabilize for 5 minutes. Following the stabilization period the selected RCP in the ruptured loop should be enabled and started. The 5 minute time delay is necessary to allow for mixing of a possible slug of water with reduced boron concentration from the ruptured SG loop. Loop 2 is the intact loop in this question making answers A and C wrong. Answer D is wrong because the delay time is much greater than 5 minute and the shocking the ruptured

SG is not the reason for the delay. Also only 1 RCP per loop is directed by the procedure.

References:

ANO-2-LP-RO-ESGTR, Revision 3, Objective 13
OP 2202.004, Steam Generator Tube Rupture, Revision 005-00-0, Step 44 and the Technical Guidance AOP 2202.004 Step 44.

Historical Comments:

QID:	0439	**Rev:**	0	**Rev Date:**	12/06/2001	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.2	**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K/A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.2						
RO Tier:	3	**RO Group:**	2.3	**RO Imp:**	2.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Knowledge of facility ALARA program.										

Question:

During the performance of a resin transfer to a cask in the train bay, a qualified Category III Radiation worker must enter a high radiation area. Which ONE of the following correctly satisfies the Radiological Protection(RP) entry requirements in addition to an approved RWP.

A. Knowledge of the dose rates in the area, obtain an appropriate radiation survey meter, and wear an electronic alarming dosimeter.

B. Always have a Radiological Protection personnel escort with an appropriate survey meter and wear an electronic alarming dosimeter.

C. Knowledge of the dose rates in the area, obtain an appropriate radiation survey meter, and wear a self reading pocket dosimeter (SRPD).

D. Always have a Radiological Protection personnel escort with an appropriate survey meter and wear a self reading pocket dosimeter (SRPD).

Answer:

A. Knowledge of the dose rates in the area, obtain an appropriate radiation survey meter, and wear an electronic alarming dosimeter.

Notes:

As long as the individual is qualified a Category III Radiation worker, he does not require a RP escort to enter the area. All RO trainees are qualified as a CAT III Radiation worker. However with out an RP escort the entering individual must be informed of the area dose rates. An alarming dosimeter is always required when entering a high radiation area. The SRPDs do not have an alarming function. This makes answer A correct and the rest wrong.

References:

ANO-S-LP-RO-RADPRO, Revision 00, Objective 10
OP 1012.017, Radiological Postings and Entry/Exit Requirements, Revision 007-00-0, Step 6.5

Historical Comments:

QID:	0441	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	4	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Loss of Vital AC Electrical Instrument Bus	**System Number**	057	**K/A:**	AA2.05						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	3.8
Description:	Ability to determine an interpret S/G pressure and level meters as they apply to the Loss of a Vital AC Instrument Bus.										

Question:

Given the following conditions:

* The plant is at 100% Power.
* The Low Steam Generator(SG) Level Bistable on PPS Channel D has been declared inoperable due to spurious trips and has been placed in Trip Channel Bypass.
* Currently there is no Low Steam Generator(SG) Level Bistable trip in on PPS Channel D.
* Power is now lost to the Low SG Level Bistable on PPS Channel B.
* I&C has been contacted to initiate troubleshooting and repair on PPS Channel B bistable.

To comply with Technical specifications and allow I&C to perform repairs, the correct sequence of actions to take would be:

A. Within 1 hour from the loss of power, remove channel D from bypass, then place channel B in bypass, then place channel D in trip.

B. Within 1 hour from the loss of power, place channel B in bypass, then place channel D in trip, then remove channel D from bypass.

C. Within 4 hours from the loss of power, remove channel D from bypass, then place channel D in trip, then place channel B in bypass.

D. Within 4 hours from the loss of power, remove channel D from bypass, then place channel B in bypass, then place channel D in trip.

Answer:

A. Within 1 hour from the loss of power, remove channel D from bypass, then place channel B in bypass, then place channel D in trip.

Notes:

Answer A is the correct sequence to prevent a plant trip. Answer B would trip the plant due to both bypasses removed when channel D is place in trip due to the interlock between the channels. Answer C would cause a plant trip due to both channels in trip when channel D is removed from bypass. Answer D is the correct sequence but would not comply with the time requirement in technical specifications.

References:

ANO-2-LP-RO-RPS, Reactor Protection System, Objective 11 and Objective 6.
Unit 2 Tech Specs 3.3.1.1 Action 2 and Action 3.

Historical Comments:

1/10/2002. Reworded Distracter C based on NRC feedback. BNC

QID:	0442	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.5	**10CFR55_43:**		**Section:**	3.6	**Type:**	Electrical				
System Title:	AC Electrical Distribution System	**System Number**	062	**K/A:**	A1.03						
RO Tier:	2	**RO Group:**	2	**RO Imp:**	2.5	**SRO Tier:**	2	**SRO Group:**	2	**SRO Imp:**	2.8

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AC Distribution System controls including the effect on instrumentation and controls of switching power supplies.

Question:

Given the following conditions:

* The plant is in Mode 3.
* RCS Tave is 505°F.
* Preparations are being made to start the 4th Reactor Coolant Pump (RCP)

When the RCP is taken to START, which ONE of the following design features is provided to prevent an electrical bus undervoltage event:

A. An undervoltage relay block on the vital 4160 volt AC busses for 25 seconds.

B. An undervoltage relay block on the vital 480 volt AC busses for 25 seconds.

C. A source undervoltage relay block on the Startup #3 for 45 seconds.

D. A 2H1/2H2 bus undervoltage lockout relay block for 45 seconds.

Answer:

B. An undervoltage relay block on the vital 480 volt AC busses for 25 seconds.

Notes:

The undervoltage protection relaying circuit is blocked on the vital 480 volt busses 2B5/6 for 25 seconds on a RCP start due to past industry experience shows that there is a potential to degrade the vital 480 volt busses low enough to auto start the EDGS. If the undervoltage relay block stays in for more than 45 seconds , then an alarm, 2B5/6 Protection Inoperative, comes in the control room
The other relays in the three distracters are not blocked thus they are wrong.

References:

ANO-2-LP-RO-ED480, 480 Volt Electrical Distribution, Obj. 7, Rev. 1
STM 2-3-2, RCPs & RCP Vibration Monitoring System, Section 1.7 Rev. 5
STM 2-32-3, 480v Distribution System, Section 2.2.2, Rev. 4
STM 2-32-2, High Voltage Electrical Distribution, Section 5.2, Rev. 6

Historical Comments:

QID:	0443	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedure/Plan	**System Number**		**K/A:**	2.4.10						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	4	**SRO Imp:**	3.1
Description:	Knowledge of Annunciator Response Procedures										

Question:

Given the following conditions:

* The plant is at full power
* Anunnciator 2K10 A-2, COLSS POWER MARGIN EXCEEDED, alarms
* The CBOR reports that Smooth Licensed Power (CV5001) reads 101.2% power

As the CRS, the correct direction that should be given to the crew is to:

A. Reduce power to less than 100% within the next Ten (10) minutes.

B. Reduce power to less than the 100% immediately.

C. Reduce power slowly to lower the 8 hour average power to < 100%.

D. Reduce power slowly to lower the 4 hour average power to < 100%.

Answer:

B. Reduce power to less than the 100% immediately.

Notes:

Annunciator 2K10 A-2 has direction that if Smoothed Licensed Power (CV5001) is greater than licensed power limit but less than licensed power limit + 1%, THEN reduce power below licensed power limit within the next 10 minutes. But if Smoothed Licensed Power (CV5001) is greater than licensed power limit + 1%, THEN reduce power below licensed power limit IMMEDIATELY. IF Smoothed Licensed Power rolling 4 hour average greater than Licensed Power Limit, THEN reduce power to prevent 8-hour average from exceeding Licensed Power Limit. In this case we are greater than 101%, then immediate action is required to reduce power below 100%. Answer C is correct and the other three distracters are incorrect.

References:

ANO-2-LP-RO-COLSS, Obj. 22, Rev. 10
2203.012J, Annunciator Corrective Action A-2, COLSS Power Margin Exceeded, Rev. 028-05-0

Historical Comments:

1/10/2002. Removed "license limit" from answer B because it cues answer - NRC feedback> BNC

QID:	0444	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	3	**Taxonomy:**	C	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.7	**10CFR55_43:**		**Section:**	4.4	**Type:**	CE EPE/APE				
System Title:	Loss of Main Feedwater	**System Number**	E06	**K/A:**	EA1.2						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.4	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.0

Description: Ability to operate and/or monitor operating behavior characteristics of the facility as they apply to Loss of Feedwater.

Question:

Given the following conditions:

* The plant has tripped from 100% power on low Steam Generator 'A' Level.
* Containment pressure is 20 psia and rising rapidly.
* Containment sump levels are rising rapidly.
* Steam Generator Pressures are 1000 psia and constant
* Steam Generator 'A' level is 20% and slowly dropping.
* Steam Generator 'B' level is 40% and slowly dropping.
* Feedwater flow to 'A' Steam Generator is very erratic.
* Hotwell level is dropping rapidly.
* All other conditions are normal.

The above conditions would indicate a:

A. Excess Steam Demand inside containment upstream of the 'A' Main Steam Isolation but downstream of the 'A' Steam flow venturi.

B. Loss of Coolant Accident inside containment in the Loop 1 RCS Hot leg on the Pressurizer Surge line.

C. Feedwater Line Break inside containment downstream of the Inlet Feedwater Check Valve, 2FW-5A.

D. Feedwater Line Break inside containment upstream of the Inlet Feedwater Check Valve, 2FW-5A.

Answer:

D. Feedwater Line Break inside containment upstream of the Inlet Feedwater Check Valve, 2FW-5A.

Notes:

Steam Generator 'A' Pressure and level would drop rapidly if an Excess Steam demand event or Feedline Break were to occur downstream of the Feedwater Line Check valve. With the constant SG pressure and slowly dropping SG level indications given (due to steaming to the Main Condenser via SDBCS), the leak is upstream of the check valve 2FW-5A. This makes answer D correct and distracters A/C wrong. Answer B is wrong because no radiation indications are rising.

References:

ANO-2-LP-RO-ESPTA, Obj. 4, Rev. 5
OP 2202.001, Standard Post Trip Actions, Step 8.C, Rev.005-00-0
STM 2-19, Main Feedwater System, Figure main feedwater system, Rev. 7

Historical Comments:

1/10/2002. Corrected typos in notes- indications based on NRC feedback. BNC

QID:	0445	**Rev:**	0	**Rev Date:**	12/07/2001	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	2	**Taxonomy:**	An	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**	43.5	**Section:**	4.2	**Type:**	Generic APE				
System Title:	Continuous Rod Withdrawal	**System Number**	001	**K/A:**	AA2.04						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	1	**SRO Group:**	1	**SRO Imp:**	4.3

Description: Ability to determine and interpret reactor power and its trend as it applies to the Continuous Rod Withdrawal.

Question:

Given the following:

* A plant startup is in progress late in core life.
* CEA Regulating Group 'P' is at 120 inches withdrawn and the reactor is CRITICAL.
* All Critical Data has been taken.
* All PPS High Log Power Trips have been bypassed.
* All CPC channels have been removed from bypass.
* Reactor Power is 2E-2%
* The CBOR selects Manual Group 'MG' CEA Mode and withdraws CEA Group 'P' to 130 inches withdrawn to commence an up power.
* The CBOR takes the CEA Mode select switch to OFF.
* The CBOR reports that Group 'P' CEA #26 is continuing to step out.

With no operator action, reactor power would ___________________ and the correct direction to give to the CBOR would be to____.

A. continue to rise without stopping; trip reactor and enter Standard Post Trip Actions.

B. rise to a higher level and stop; select Manual Individual (MI) CEA Mode and Insert CEA #26.

C. rise to a higher level and stop; trip reactor and enter Standard Post Trip Actions.

D. continue to rise without stopping; select Manual Individual (MI) CEA Mode and Insert CEA #26.

Answer:

C. rise to a higher level and stop; trip reactor and enter Standard Post Trip Actions.

Notes:

Reactor power would start to rise but late in core life, a definite negative moderator temperature coefficient would be offset by the positive reactivity added by the one withdrawing CEA and level reactor power. The CEA would only go to 150 inches withdrawn and be stopped by the upper electrical limit. The correct action to take in the CEA Malfunction AOP step 2 is to verify all CEA movement stopped and if not, the contingency is to trip the reactor. The correct combination is answer C and the other distracters are wrong.

References:

ANO-2-LP-SRO-AOP, Revision 7, Objective 6
OP 2203.003, CEA Malfunction, Revision 014-02-0, Step 2 and its associated Technical Guidance.
STM 2-02, CEDMCS, Revision 8, Section 1.2
General Physics Reactor Theory Chapter 8, Reactor Operational Physics, Revision 2, Intermediate Range Operations.

Historical Comments:

QID:	0446	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	No	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	S	**Difficulty:**	3	**Taxonomy:**	Ap	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.10	**10CFR55_43:**	43.5	**Section:**	2.4	**Type:**	Generic K/A				
System Title:	Emergency Procedures/Plan	**System Number**		**K/A:**	2.4.11						
RO Tier:		**RO Group:**		**RO Imp:**		**SRO Tier:**	3	**SRO Group:**	4	**SRO Imp:**	3.6
Description:	Knowledge of abnormal condition procedures.										

Question:

Given the following conditions:

* The plant is in Mode 5 preparing to conduct refueling
* The RCS is being drained to mid-loop
* The Low Pressure Injection Pump (LPSI) for the inservice Shutdown Cooling (SDC) train starts having abnormal flow oscillations and a reduction in discharge pressure.
* Current SDC flow is oscillating between 200 and 400 gpm.
* The inservice LPSI pump amps are oscillating between 25 and 30 amps.

Which ONE of the following is the correct response to these conditions?

A. Enter the Loss of SDC procedure, contact Instrument and Control technicians to investigate indicators for possible failure and initiate a MAI.

B. Enter the Loss of SDC procedure, secure the LPSI Pump, and go to the Lower Mode Functional Recovery procedure.

C. Enter the Lower Mode Functional Recovery procedure and leave the running LPSI pump in service for RCS heat removal concerns.

D. Enter the Lower Mode Functional Recovery procedure, secure the LPSI Pump, then go to the Loss of SDC procedure.

Answer:

B. Enter the Loss of SDC procedure, secure the LPSI Pump, and go to the Lower Mode Functional Recovery procedure.

Notes:

Actions to take for indications of a cavitating LPSI are found in the Loss of SDC procedure so this should be entered first. This procedure directs securing of the running LPSI pump if flow is less than 500 gpm and then directs that the Lower Mode Recovery procedure be implemented. Distracter A is incorrect because of more than one indication shows an actual malfunction occurring. Distracter C is incorrect because leaving the pump running could air bind both trains of SDC causing a longer recovery time. Distracter D is incorrect because of the reversal of procedures of the correct answer B.

References:

ANO-2-LP-SRO-AOP, Revision 7, Objective 32
OP 2203.029, Loss of SDC, Revision 010-03-0, Step 8.0 and its associated technical guidance.

Historical Comments:

1/10/2002. This question was generated to replace QID 428 based on feedback from the NRC that QID 428 was too similar to QID 409. BNC

QID:	0447	**Rev:**	0	**Rev Date:**	01/10/2002	**RO Select:**	Yes	**SRO Select:**	No	**Points:**	1.00
Lic Level:	R	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:		**10CFR55_43:**		**Section:**	2.1	**Type:**	Generic K/A				
System Title:	Conduct of Operations	**System Number**		**K/A:**	2.1.17						
RO Tier:	3	**RO Group:**	2.1	**RO Imp:**	3.5	**SRO Tier:**		**SRO Group:**		**SRO Imp:**	
Description:	Ability to make accurate, clear and concise verbal reports.										

Question:

Given the following:

* The CRS states to the crew "The Bravo Charging Pump is the lead Charging Pump"
* The CRS directs the CBOR to "Make the Alpha Charging Pump the lead pump"

In accordance with OPS Directive COPD-001, Operations Standards and Expectations, which ONE of the following would be the correct CBOR communication response and action to this direction?

A. "Understand make 2P36A the lead pump"; perform the action; inform the CRS when action is complete; update the status board.

B. "Understand make 2P36A the lead Pump"; wait for CRS to acknowledge; perform the action; update the status board.

C. "Understand make Alpha Charging Pump the lead pump"; wait for CRS to acknowledge; perform the action; inform the CRS when action is complete.

D. "Understand make Alpha Charging Pump the lead pump"; perform the action; inform the CRS when action is complete.

Answer:

C. "Understand make Alpha Charging Pump the lead pump"; wait for CRS to acknowledge; perform the action; inform the CRS when action is complete.

Notes:

A proper repeat back of a given direction would include the noun name of the component using the phonetic alphabet. After the repeat back of the direction, closure of the third leg is required prior to taking action. All actions taken by the board operator due to direction given by the CRS require a report back to the CRS when complete. Distracter A does not use the noun name/phonetic alphabet nor does it include the third leg of the communication. Distracter B does not use the noun name/phonetic alphabet and the CRS is not informed when action completed. Distracter D does not include the third leg of the communication/direction given.

References:

ANO-S-LP-RO-COMM, Revision 3, Objective 1.1
COPD001, Common Operations Directive on OPS Expectation and standards, Revision 7, Step 9.0.

Historical Comments:

1/10/2002. This is a new question to replace QID 395 based on feedback from the NRC that QID 395 was not a license level question. BNC

QID:	0448	**Rev:**	1	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.1 & 41.8	**10CFR55_43:**		**Section:**	4.4	**Type:**	Generic EPE/APE				
System Title:	CE Functional Recovery	**System Number**	E09	**K/A:**	EK1.2						
RO Tier:	1	**RO Group:**	2	**RO Imp:**	3.2	**SRO Tier:**	1	**SRO Group:**	2	**SRO Imp:**	4.0

Description: Knowledge of the operational implications of normal, abnormal and emergency procedures associated with Functional Recovery.

Question:

Given the following:

* During the use of the Functional Recovery Procedure, an assessment of safety functions has been completed and the RCS Inventory safety function is not meeting its acceptance criteria.
* All other safety function acceptance criteria is being met.

RCS Inventory is considered a __________________ safety function and should be addressed ____.

A. Challenged; after Reactivity Control but before Containment Isolation.

B. Jeopardized; after Reactivity Control but before Containment Isolation.

C. Jeopardized; prior to all the other safety functions.

D. Challenged; prior to all the other safety functions.

Answer:

C. Jeopardized; prior to all the other safety functions.

Notes:

Answer B is the correct definition of the stem in accordance with the EOP users guide definition 4.34.9. Challenged means the acceptance criteria is met but trending the wrong way and action is required to prevent loss of the safety function so distracter A is wrong (definition 4.34.3). Impacted is the term used to describe a low level in a steam generator requiring a slow refill so distracter C is wrong(definition 4.34.7). Disturbed is not a term used in the functional recovery procedure so distracter D is wrong.

References:

ANO-2-LP-RO-EFRP, Revision 2, Objective 3
OP 1015.021, ANO-2 EOP/AOP Users Guide, Revision 004-02-0, Steps 4.34.3, 4.34.7, & 4.34.9, and Attachment A, Safety Function Hierarchy.

Historical Comments:

1/10/2002. This question was added to replace QID 340 which was too similar to QID 418 based on NRC feedback. BNC. 1/17/2002- added actions as to when the safety function should be addressed after assessment based on NRC feedback.

QID:	0449	**Rev:**	0	**Rev Date:**	01/17/2002	**RO Select:**	Yes	**SRO Select:**	Yes	**Points:**	1.00
Lic Level:	RS	**Difficulty:**	2	**Taxonomy:**	K	**Source:**	New	**Originator:**	Coble		
10CFR55_41:	41.12	**10CFR55_43:**	43.4	**Section:**	2.3	**Type:**	Generic K&A				
System Title:	Radiation Control	**System Number**		**K/A:**	2.3.1						
RO Tier:	2	**RO Group:**	1	**RO Imp:**	2.6	**SRO Tier:**	2	**SRO Group:**	1	**SRO Imp:**	3.0
Description:	Knowledge of 10CFR20 and related facility radiation control requirements.										

Question:

Given the following:

* The plant is in mode 5 for a refueling outage.
* Steam Generator (SG) Nozzle Dam placement is in progress.
* During placement of hot leg nozzle dams on the 'A' SG, an individual received a Shallow Dose Equivalent (SDE) to his left hand of 25 Rem.
* The local dose rate in the 'B' Steam Generator hot leg nozzle dam area is 50 Rem per hour.
* This same individual is tasked with installing the 'B' SG hot leg nozzle dam.

How long could this individual keep his left hand in the 'B' SG hot leg nozzle dam area and not exceed his SDE administrative extremity limit?

A. 18 minutes

B. 30 minutes

C. 42 minutes

D. 60 minutes

Answer:

A. 18 minutes

Notes:

A is the correct answer based on .3 hours (18 minutes) X 50 Rem per hour = 15 Rem + 25 REM = 40 REM which is the limit. Distracter B is wrong because it would add up to the Federal Limit of 50 REM. Distracter C is incorrect because it would add up to 60 REM. Distracter D is wrong because it would add up to 75 REM which is the lifetime lens dose limit for a planned special exposure.

References:

1012.021, Rev 004-02-0, Steps 6.1.1 and 6.2.2 (Exposure Limits and Controls)
ANO-S-LP-RO-RADPRO, Rev 00, Objective 14.0

Historical Comments:

ATTACHMENT 1 - TEMPORARY ALTERATION SCREENING

Temporary alteration identification guidance should be incorporated into the Work Management Process. Expectations for Operations, Maintenance, Planners and System Engineering support for the Work Management Process include the initial identification of potential temporary alterations during the job planning stages. This screening criteria may be utilized during the workweek reviews.

The screening questions are:

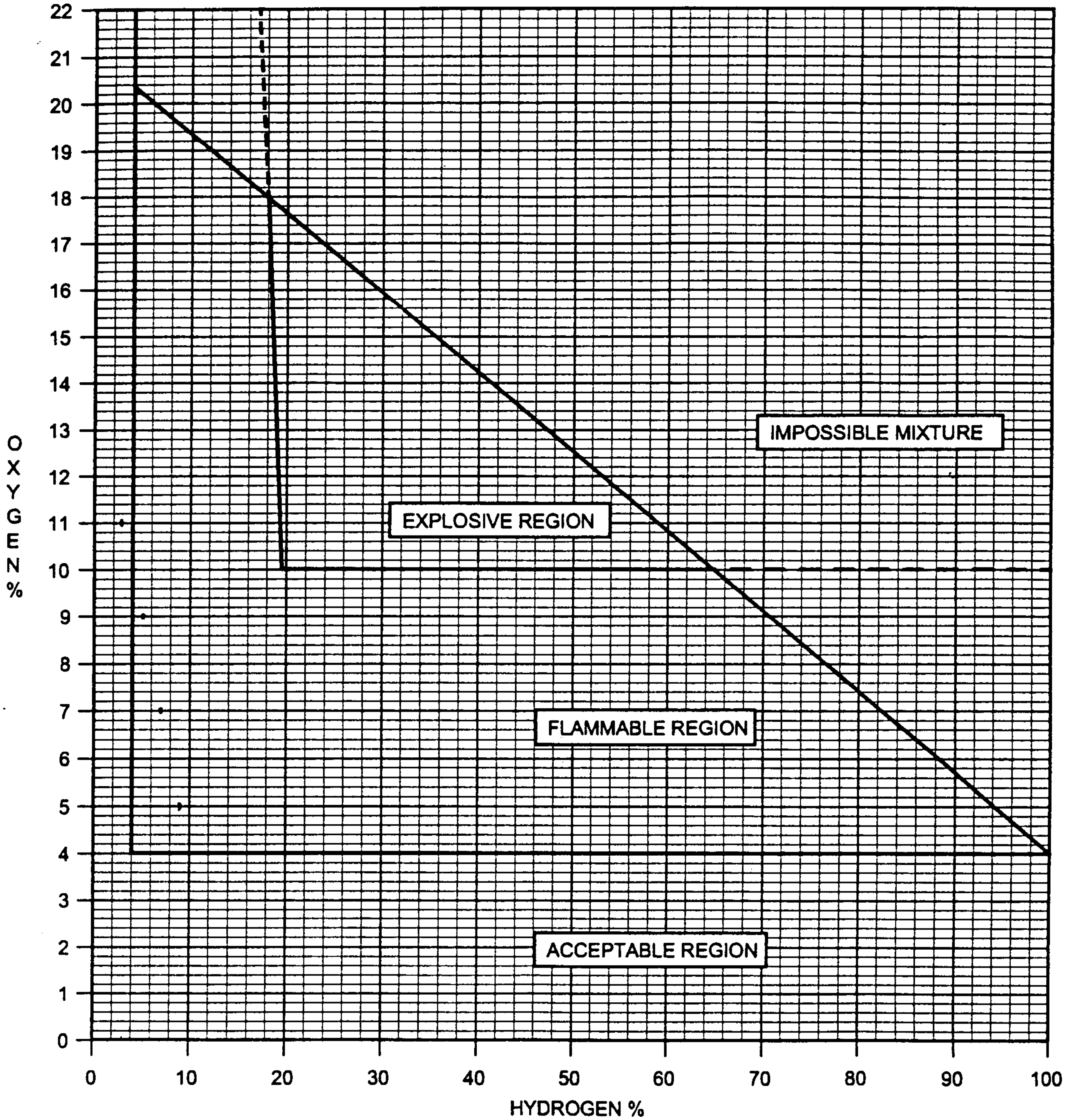
1. Is the SSC affected by this activity [or associated SSC] in service, in standby or required to be operable?
2. Does the intended activity temporarily modify the SSC configuration so that it deviates from the approved drawings or other design documents or changes the design function of the SSC?
3. Would a design change or equivalency evaluation be required to make the altered configuration permanent?

If the answers to ALL three of these questions are "YES",

THE ACTIVITY IS A POTENTIAL TEMPORARY ALTERATION AND ENGINEERING ASSISTANCE MAY BE OBTAINED.

Proceed to TEMPORARY ALTERATION DETERMINATION GUIDELINES [ATTACHMENT 2]

ATTACHMENT A
H_2/O_2 CONCENTRATIONS



ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE
TABLE 1

COMPONENT	UNIT 1 Tech Spec	UNIT 2 Tech Spec
VSF-9, 2VSF-9	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
2VUC-27A Fan or Heater	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
2VUC-27B Fan or Heater	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
2VE-1A, 2VE-1B	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
CV-7905, CV-7907	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
2UCD-8683, 2PCD-8685	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
QS-7905, QS-7907	None	None
2XSH-8740A/B, 2XSH-8741A/B	None	None
Chlorine Monitors	TRM 3.5.1.10 (Note 2)	TRM 3.3.3.7 (Note 2)
2RITS-8001A OR 2RITS-8001B	TS 3.5.1.13, Table 3.5.1-1 (Notes 1, 3)	TS 3.3.3.1, Table 3.3-6 (Notes 1, 3)
2RITS-8750-1A OR 2RITS-8750-1B	TS 3.5.1.13, Table 3.5.1-1 (Notes 1, 3)	TS 3.3.3.1, Table 3.3-6 (Notes 1, 3)

Note 1: Unit 1 TS 3.9.1, 3.9.2, and 3.5.1.13 require operability whenever the RCS is above cold shutdown or during handling of irradiated fuel.
Unit 2 TS 3.3.3.1, Table 3.3-6 and TS 3.7.6.1 require operability in Mode 1, 2, 3, and 4 and during handling of irradiated fuel. With one train of Control Room Emergency AC system inoperable, Unit 1 TS 3.9.1.2 and Unit 2 TS 3.7.6.1, Action a (if in modes 1-4) and/or Action d (if handling irradiated fuel) apply. With one train of Control Room Emergency Ventilation System inoperable, Unit 1 TS 3.9.2.2 and Unit 2 TS 3.7.6.1, Action b (if in modes 1-4) and/or Action e (if handling irradiated fuel) apply. With one train of Control Room Emergency Ventilation System AND one train of Control Room Emergency AC system inoperable, Unit 2 TS 3.7.6.1, Action c (if in modes 1-4) and/or Action f (if handling irradiated fuel) apply. During handling of irradiated fuel, with both trains of Control Room Emergency AC System inoperable OR both trains of Control Room Emergency Ventilation System inoperable Unit 2 TS 3.7.6.1 Action g applies.

Note 2: Chlorine monitors, per TRMs, are required to be operable above cold shutdown for Unit 1 and in Mode 1, 2, 3, and 4 for Unit 2. Chlorine monitors are NOT required to be operable during handling of irradiated fuel.

Note 3: Two of the four radiation monitors have to be operable to satisfy Unit 1 and Unit 2 Tech Specs. The two must be comprised of the following: one Unit 1 monitors (2RITS-8001A or 2RITS-8001B) and one Unit 2 monitor (2RITS-8750-1A or 2RITS-8750-1B). If both monitors are inoperable on one Unit, and any monitor is operable on the other Unit, then both units will enter the appropriate TS action statement (Unit 1 TS Table 3.5.1-1, Action 18 and Unit 2 TS Table 3.3-6, Action 20). If no monitors are operable on either unit (i.e., 2RITS-8001A, 2RITS-8001B, 2RITS-8750-1A, and 2RITS-8750-1B are inoperable), then Unit 1 TS Table 3.5.1-1, Action 17 and Unit 2 TS Table 3.3-6 Action 17 apply.

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE

1.0 The operability of the Control Room Emergency Ventilation and Air Conditioning Systems are required when either Unit is above cold shutdown and during handling of irradiated fuel. Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

2.0 When the Emergency Cooling Pond is NOT available to Unit 2 with both Emergency Air Conditioning Systems aligned to Unit 2, then Unit 1 Tech Spec 3.9.1 is NOT satisfied due to assumed single failure of the Dardanelle Dam. (LIC-97-049)

3.0 When an Emergency Control Room Air Conditioning System (2VUC-27A/2VUC-27B) is powered from Unit 1, then Unit 2 Tech Spec 3.7.6.1 is NOT satisfied because components can not be started from the Control Room. The system remains operable for Unit 1.

4.0 When power is supplied to 2VUC-27A or B through 2B5/2B6 crosstie, then: 1) Unit 1 is NOT in a Tech Spec action. Power independence is not required by Unit 1 licensing bases documents. 2) Unit 2 is in a 30-day time clock (TS 3.7.6.1.a). One of the Air Conditioning units (2VUC-27A or 2VUC-27B) shall be declared inoperable. Because the fans do not have to start automatically, as long as the capability to supply power to them via the cross-tied bus is available, one fan can be considered operable for Unit 2.

5.0 Per Licensing memo LIC-092-49, both Emergency Ventilation systems (VSF-9 and 2VSF-9) may be powered from Unit 1 on the same electrical bus without entering the Unit 1 Tech Spec. Unit 2 shall enter Tech Spec 3.7.6.1, Action b.

6.0 If Unit 1 is above CSD and the normal or emergency power supply to any TS component is removed from service, then Unit 1 applies TS 3.0.5 and Unit 2 declares the associated component inoperable and enters the applicable TS LCO.

7.0 If Unit 2 is in Mode 1, 2, 3, or 4 and the normal or emergency power supply to any TS component is removed from service, then Unit 2 applies TS 3.0.5 and Unit 1 declares the associated component inoperable and enters the applicable TS LCO. This does not apply to 2VSF-9 that may be supplied with normal and emergency power per Table 2 of this attachment.

8.0 Tech Spec 3.0.5 does NOT apply for Unit 1 below CSD or for Unit 2 in Mode 5 or 6. When irradiated fuel movement is in progress when TS 3.0.5 does NOT apply, the normal and emergency power supply for each component is required to be operable.

9.0 If during handling of irradiated fuel any component becomes inoperable, fuel handling may continue providing the appropriate action statement is entered. If the time clock expires prior to repairing the component fuel handling activities shall be stopped.

10.0 Handling of irradiated fuel is considered any of the following:

- Irradiated fuel movement in the fuel pool area in any mode or defueled.
- Irradiated fuel movement in either Unit's Reactor Building.

CEA shuffle, new fuel movement, and sealed cask movements (i.e. when both lids are welded in place) are NOT considered part of irradiated fuel handling.

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE

TABLE 2

Equipment	Normal Power	Emergency Power	Alternate Normal Power	Alternate Emergency Power
VSF-9	B5553	1DG1 or 1DG2 to B55	2B64-D1	2DG2 to 2B64-D1
2VSF-9	2B54-C3	2DG1 to 2B54	Note 1	Note 1
2VUC-27A Fan	2B54-B1	2DG1 to 2B54	B64A-4A	1DG2 to B64A-4A
2VUC-27A Heater	2B54-D2	2DG1 to 2B54	B64A-2A	1DG2 to B64A-2A
2VUC-27B Fan	2B61-J4	2DG2 to 2B61	B64B-5A	1DG2 to B64B-5A
2VUC-27B Heater	2B61-J5	2DG2 to 2B61	B64B-3A	1DG2 to B64B-3A
2VE-1A	2B52-D5	2DG1 to 2B52	B64A-5A	1DG2 to B64A-5A
2VE-1B	2B63-K6	2DG2 to 2B63	B64B-6A	1DG2 to B64B-6A

Note 1:

Alternate Normal power for 2VSF-9 may be any one of the following:

1) B5666

2) Aligned to an operable Unit 2 EDG. 2B5 and 2B6 may be cross-tied.

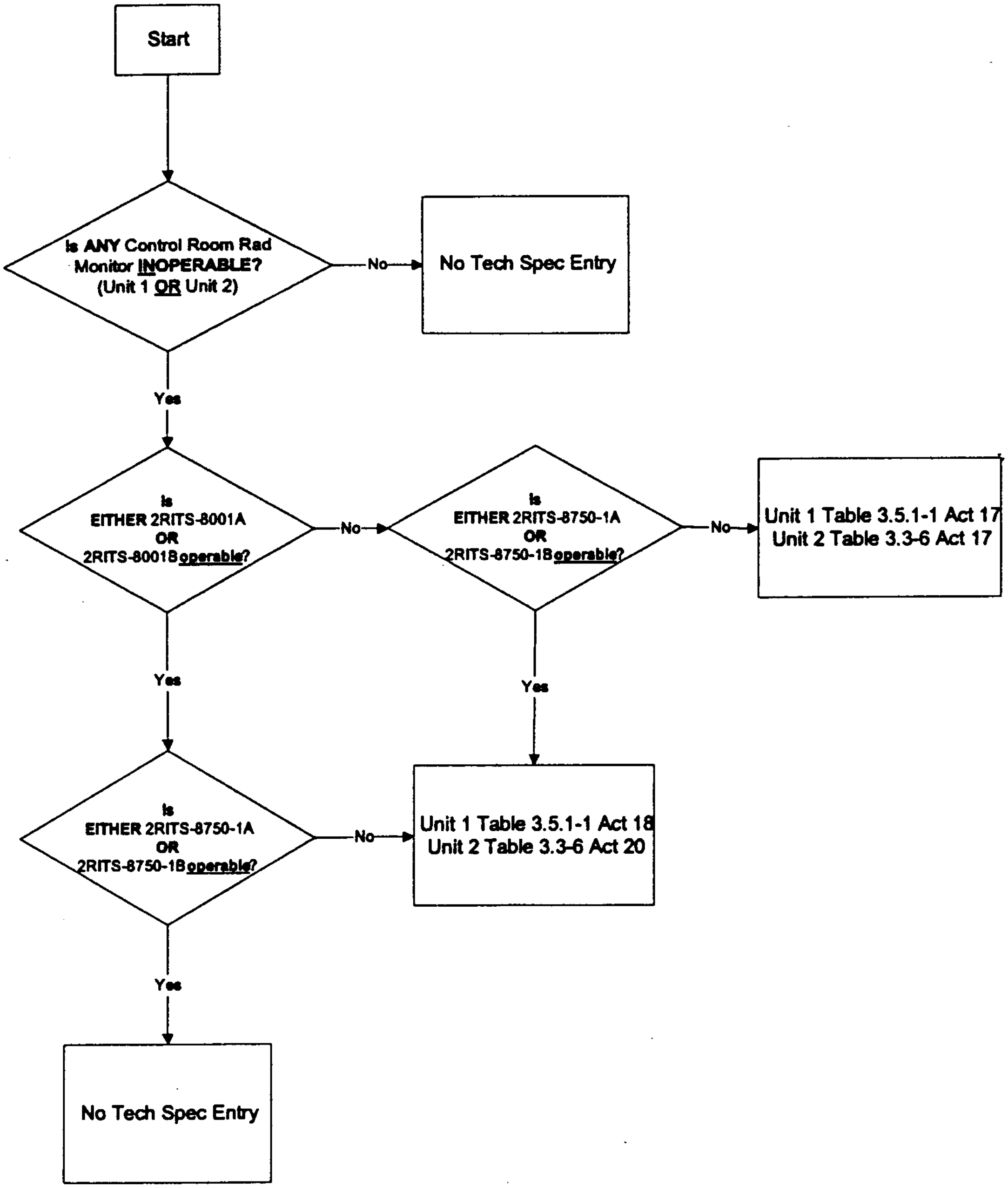
Alternate Emergency power for 2VSF-9 may be any one of the following:

1) 1DG1 or 1DG2 capable of being aligned to B5666

2) Aligned or capable of being aligned to an operable Unit 2 EDG. For Unit 1, 2EDG1 or 2EDG2 may serve as the emergency power supply. It is only necessary that 2EDG2 be capable of being aligned to 2VSF-9 (i.e. 2B5 and 2B6 are not required to be physically cross-tied for an operable 2EDG2 to serve as the 2VSF-9 emergency power source for Unit 1).

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE



INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONTIORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	$10^{-4} - 10^{1}$ R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	$1 - 10^{7}$ R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	5 & 6	$\leq$ 2 x background	$10 - 10^{6}$ cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	$\leq$ 2 x background	$10 - 10^{6}$ cpm	17, 20
c. Main Steam Line Radiation Monitors	1/Steam Line	1, 2, 3, & 4	Not Applicable	$10^{-1} - 10^{4}$ mR/hr	19

Note 1 - With fuel in the spent fuel pool or building.
Note 2 - MODES 1, 2, 3, 4, and during handling of irradiated fuel.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If performing CORE ALTERATIONS or moving irradiated fuel within the reactor building, secure the containment purge system or suspend CORE ALTERATIONS and movement of irradiated fuel within the reactor building.

b. If a containment PURGE is in progress, secure the containment purge system.

c. If continuously ventilating, verify the SPING monitor operable or perform the ACTIONS of 3.3.3.9, or secure the containment purge system.

ACTION 17 - With no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 18 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

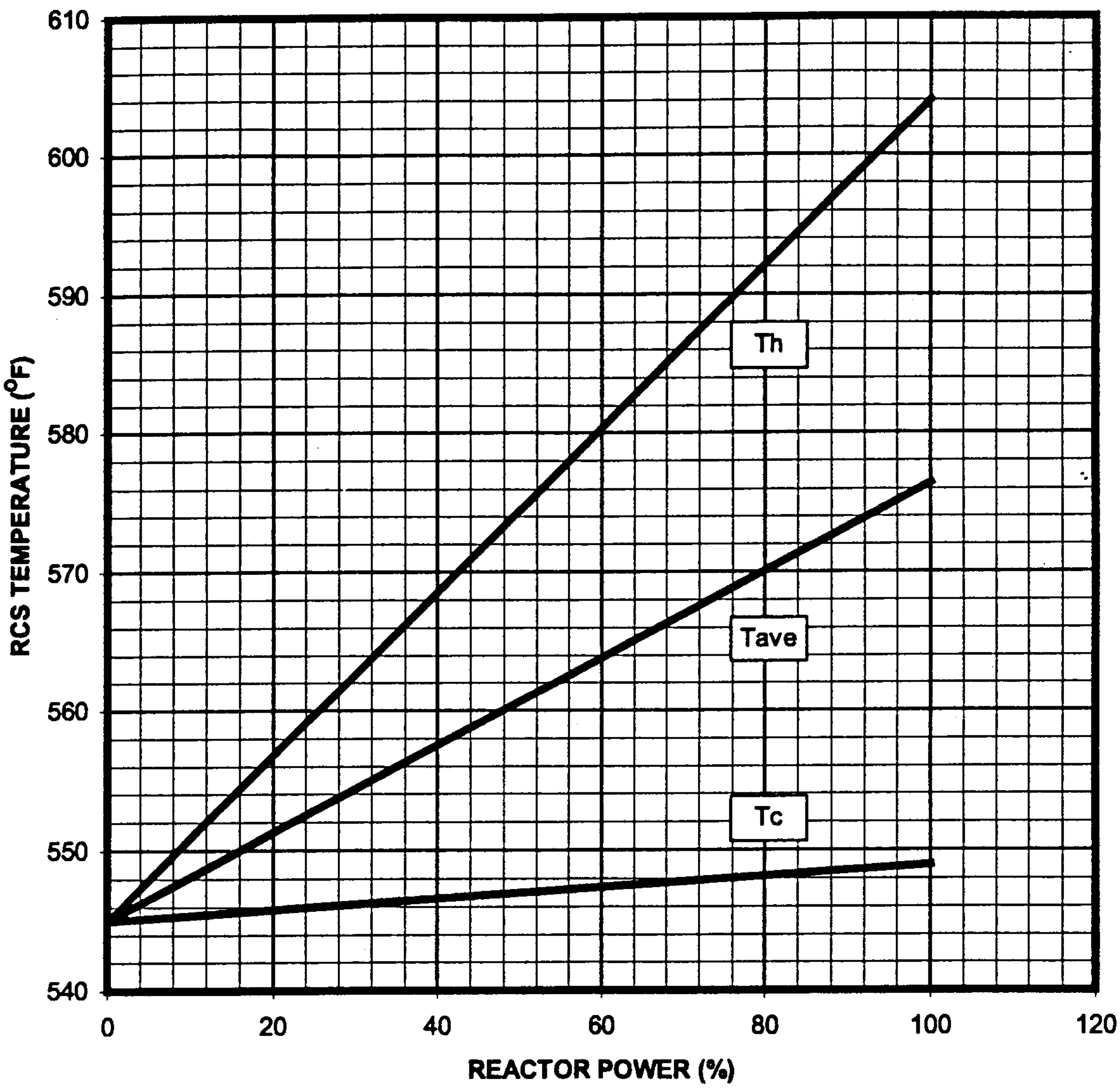
ACTION 20 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or within the next 6 hours initiate and maintain the control room emergency ventilation system in the recirculation mode of operation.

ATTACHMENT C

PAGE 1 OF 1

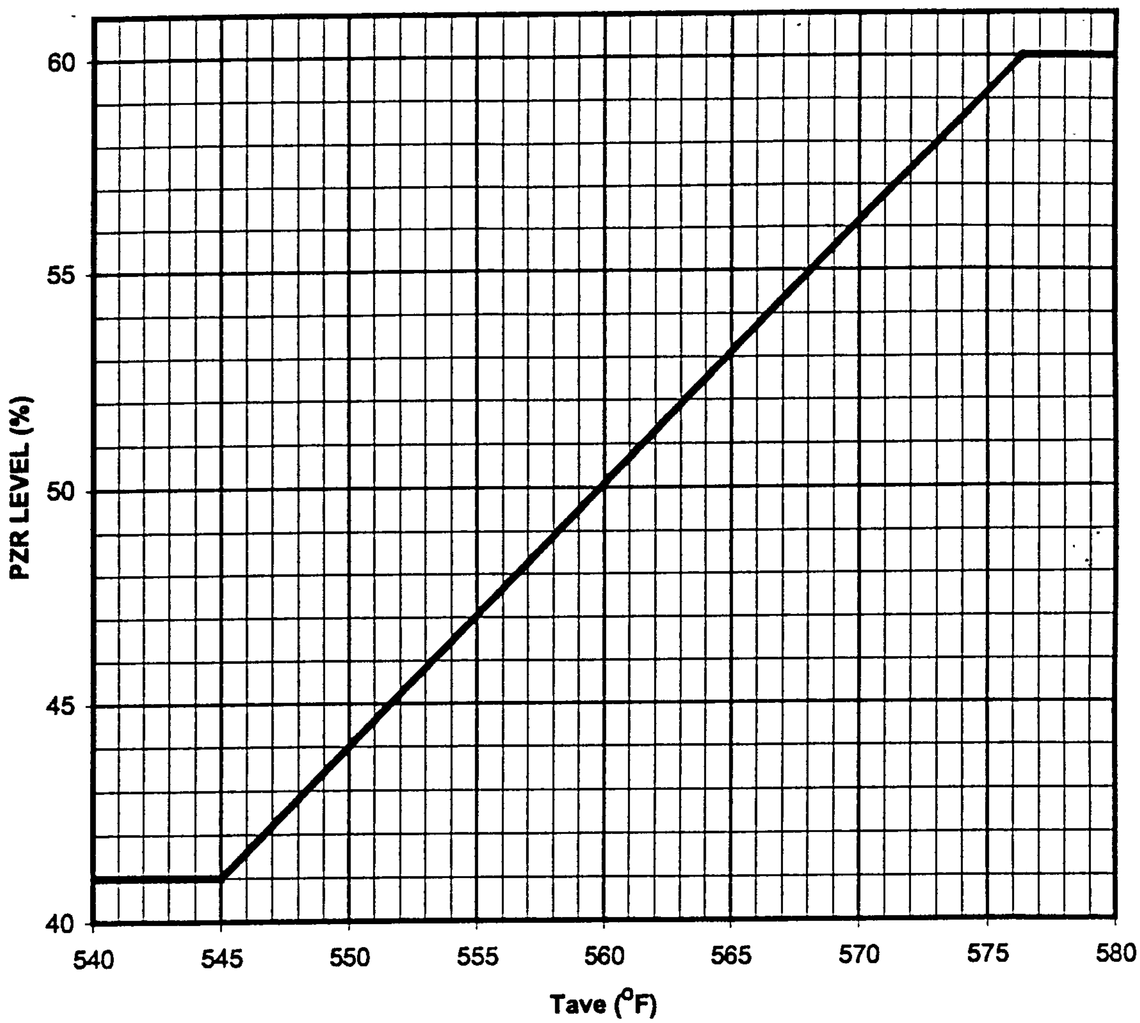
RCS TEMPERATURE VS REACTOR POWER

This temperature profile represents the desired trend for RCS temperature vs. Reactor power levels. The actual values at near full power may vary.



ATTACHMENT E

PRESSURIZER LEVEL PROGRAM



3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR limit shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and at least one CEAC is operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the value specified in the CORE OPERATING LIMITS REPORT (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and at least one CEAC is operable); or

d. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

   1. Restore the DNBR to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the DNBR limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the DNBR to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

6) 3/4.2.1 - LINEAR HEAT RATE

With COLSS out of service, the linear heat rate shall be maintained ≤ 13.5 kW/ft.

7) 3.2.3 - AZIMUTHAL POWER TILT- T_q

The measured AZIMUTHAL POWER TILT shall be maintained ≤ 0.03.

8) 3/4.2.4 - DNBR MARGIN

The DNBR limit shall be maintained by one of the following methods:

a) With COLSS in service and neither CEAC operable - Maintain COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13%.

b) With COLSS out of service and at least one CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 4, using any operable CPC channel.

c) With COLSS out of service and neither CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 5, using any operable CPC channel.

9) 3.2.7 - AXIAL SHAPE INDEX

The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a) COLSS IN SERVICE
$-0.27 \leq \text{ASI} \leq +0.27$

b) COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

FIGURE 4

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, CEAC OPERABLE)

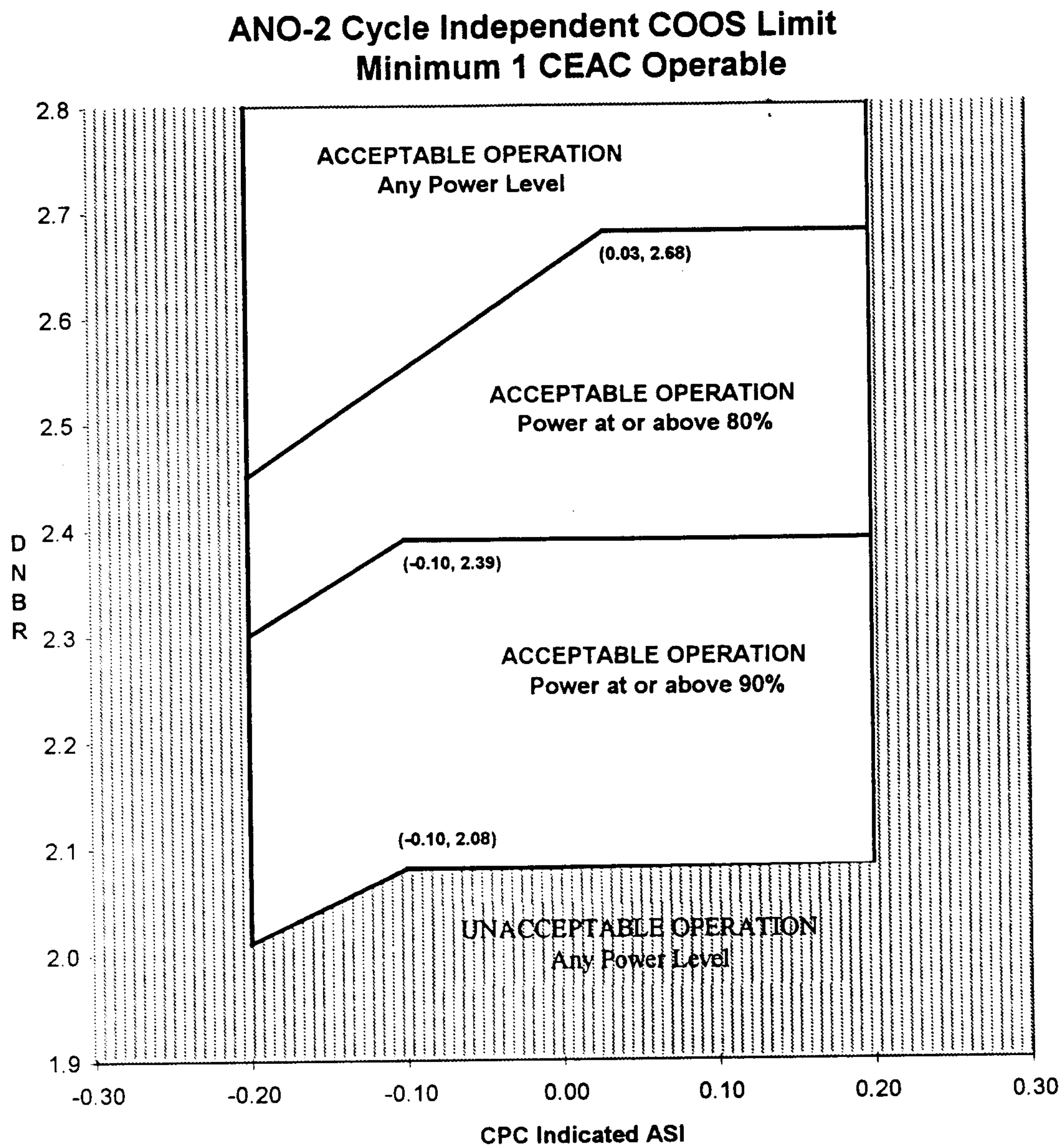
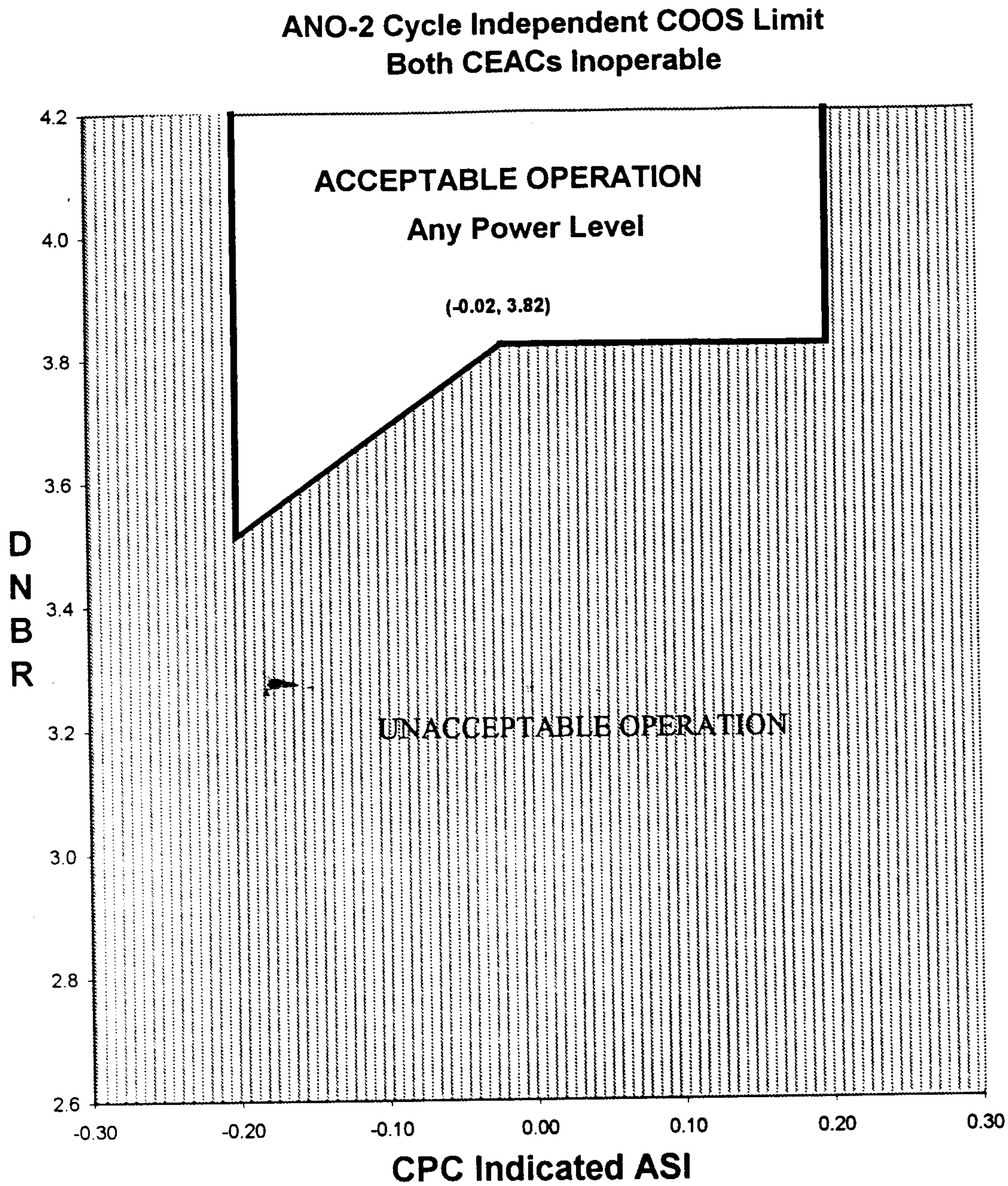


FIGURE 5

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, BOTH CEACS INOPERABLE)



ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE
TABLE 1

COMPONENT	UNIT 1 Tech Spec	UNIT 2 Tech Spec
VSF-9, 2VSF-9	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
2VUC-27A Fan or Heater	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
2VUC-27B Fan or Heater	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
2VE-1A, 2VE-1B	TS 3.9.1 (Note 1)	TS 3.7.6.1 (Note 1)
CV-7905, CV-7907	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
2UCD-8683, 2PCD-8685	TS 3.9.2 (Note 1)	TS 3.7.6.1 (Note 1)
QS-7905, QS-7907	None	None
2XSH-8740A/B, 2XSH-8741A/B	None	None
Chlorine Monitors	TRM 3.5.1.10 (Note 2)	TRM 3.3.3.7 (Note 2)
2RITS-8001A OR 2RITS-8001B	TS 3.5.1.13, Table 3.5.1-1 (Notes 1, 3)	TS 3.3.3.1, Table 3.3-6 (Notes 1, 3)
2RITS-8750-1A OR 2RITS-8750-1B	TS 3.5.1.13, Table 3.5.1-1 (Notes 1, 3)	TS 3.3.3.1, Table 3.3-6 (Notes 1, 3)

Note 1: Unit 1 TS 3.9.1, 3.9.2, and 3.5.1.13 require operability whenever the RCS is above cold shutdown or during handling of irradiated fuel. Unit 2 TS 3.3.3.1, Table 3.3-6 and TS 3.7.6.1 require operability in Mode 1, 2, 3, and 4 and during handling of irradiated fuel. With one train of Control Room Emergency AC system inoperable, Unit 1 TS 3.9.1.2 and Unit 2 TS 3.7.6.1, Action a (if in modes 1-4) and/or Action d (if handling irradiated fuel) apply. With one train of Control Room Emergency Ventilation System inoperable, Unit 1 TS 3.9.2.2 and Unit 2 TS 3.7.6.1, Action b (if in modes 1-4) and/or Action e (if handling irradiated fuel) apply. With one train of Control Room Emergency Ventilation System AND one train of Control Room Emergency AC system inoperable, Unit 2 TS 3.7.6.1, Action c (if in modes 1-4) and/or Action f (if handling irradiated fuel) apply. During handling of irradiated fuel, with both trains of Control Room Emergency AC System inoperable OR both trains of Control Room Emergency Ventilation System inoperable Unit 2 TS 3.7.6.1 Action g applies.

Note 2: Chlorine monitors, per TRMs, are required to be operable above cold shutdown for Unit 1 and in Mode 1, 2, 3, and 4 for Unit 2. Chlorine monitors are NOT required to be operable during handling of irradiated fuel.

Note 3: Two of the four radiation monitors have to be operable to satisfy Unit 1 and Unit 2 Tech Specs. The two must be comprised of the following: one Unit 1 monitors (2RITS-8001A or 2RITS-8001B) and one Unit 2 monitor (2RITS-8750-1A or 2RITS-8750-1B). If both monitors are inoperable on one Unit, and any monitor is operable on the other Unit, then both units will enter the appropriate TS action statement (Unit 1 TS Table 3.5.1-1, Action 18 and Unit 2 TS Table 3.3-6, Action 20). If no monitors are operable on either unit (i.e., 2RITS-8001A, 2RITS-8001B, 2RITS-8750-1A, and 2RITS-8750-1B are inoperable), then Unit 1 TS Table 3.5.1-1, Action 17 and Unit 2 TS Table 3.3-6 Action 17 apply.

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE

1.0 The operability of the Control Room Emergency Ventilation and Air Conditioning Systems are required when either Unit is above cold shutdown and during handling of irradiated fuel. Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

2.0 When the Emergency Cooling Pond is NOT available to Unit 2 with both Emergency Air Conditioning Systems aligned to Unit 2, then Unit 1 Tech Spec 3.9.1 is NOT satisfied due to assumed single failure of the Dardanelle Dam. (LIC-97-049)

3.0 When an Emergency Control Room Air Conditioning System (2VUC-27A/2VUC-27B) is powered from Unit 1, then Unit 2 Tech Spec 3.7.6.1 is NOT satisfied because components can not be started from the Control Room. The system remains operable for Unit 1.

4.0 When power is supplied to 2VUC-27A or B through 2B5/2B6 crosstie, then: 1) Unit 1 is NOT in a Tech Spec action. Power independence is not required by Unit 1 licensing bases documents. 2) Unit 2 is in a 30-day time clock (TS 3.7.6.1.a). One of the Air Conditioning units (2VUC-27A or 2VUC-27B) shall be declared inoperable. Because the fans do not have to start automatically, as long as the capability to supply power to them via the cross-tied bus is available, one fan can be considered operable for Unit 2.

5.0 Per Licensing memo LIC-092-49, both Emergency Ventilation systems (VSF-9 and 2VSF-9) may be powered from Unit 1 on the same electrical bus without entering the Unit 1 Tech Spec. Unit 2 shall enter Tech Spec 3.7.6.1, Action b.

6.0 If Unit 1 is above CSD and the normal or emergency power supply to any TS component is removed from service, then Unit 1 applies TS 3.0.5 and Unit 2 declares the associated component inoperable and enters the applicable TS LCO.

7.0 If Unit 2 is in Mode 1, 2, 3, or 4 and the normal or emergency power supply to any TS component is removed from service, then Unit 2 applies TS 3.0.5 and Unit 1 declares the associated component inoperable and enters the applicable TS LCO. This does not apply to 2VSF-9 that may be supplied with normal and emergency power per Table 2 of this attachment.

8.0 Tech Spec 3.0.5 does NOT apply for Unit 1 below CSD or for Unit 2 in Mode 5 or 6. When irradiated fuel movement is in progress when TS 3.0.5 does NOT apply, the normal and emergency power supply for each component is required to be operable.

9.0 If during handling of irradiated fuel any component becomes inoperable, fuel handling may continue providing the appropriate action statement is entered. If the time clock expires prior to repairing the component fuel handling activities shall be stopped.

10.0 Handling of irradiated fuel is considered any of the following:

- Irradiated fuel movement in the fuel pool area in any mode or defueled.
- Irradiated fuel movement in either Unit's Reactor Building.

CEA shuffle, new fuel movement, and sealed cask movements (i.e. when both lids are welded in place) are NOT considered part of irradiated fuel handling.

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE

TABLE 2

Equipment	Normal Power	Emergency Power	Alternate Normal Power	Alternate Emergency Power
VSF-9	B5553	1DG1 or 1DG2 to B55	2B64-D1	2DG2 to 2B64-D1
2VSF-9	2B54-C3	2DG1 to 2B54	Note 1	Note 1
2VUC-27A Fan	2B54-B1	2DG1 to 2B54	B64A-4A	1DG2 to B64A-4A
2VUC-27A Heater	2B54-D2	2DG1 to 2B54	B64A-2A	1DG2 to B64A-2A
2VUC-27B Fan	2B61-J4	2DG2 to 2B61	B64B-5A	1DG2 to B64B-5A
2VUC-27B Heater	2B61-J5	2DG2 to 2B61	B64B-3A	1DG2 to B64B-3A
2VE-1A	2B52-D5	2DG1 to 2B52	B64A-5A	1DG2 to B64A-5A
2VE-1B	2B63-K6	2DG2 to 2B63	B64B-6A	1DG2 to B64B-6A

Note 1:

Alternate Normal power for 2VSF-9 may be any one of the following:

1) B5666

2) Aligned to an operable Unit 2 EDG. 2B5 and 2B6 may be cross-tied.

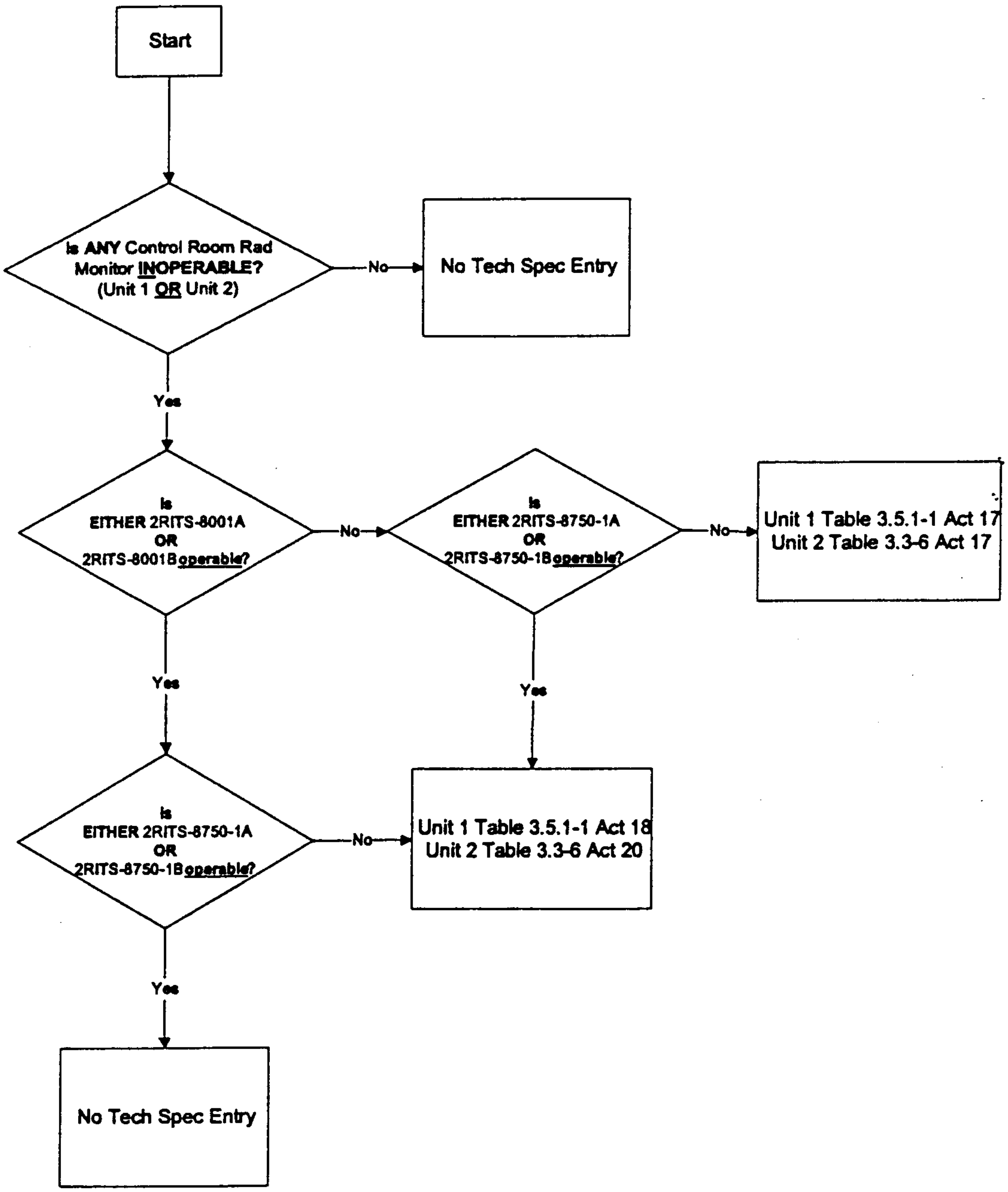
Alternate Emergency power for 2VSF-9 may be any one of the following:

1) 1DG1 or 1DG2 capable of being aligned to B5666

2) Aligned or capable of being aligned to an operable Unit 2 EDG. For Unit 1, 2EDG1 or 2EDG2 may serve as the emergency power supply. It is only necessary that 2EDG2 be capable of being aligned to 2VSF-9 (i.e. 2B5 and 2B6 are not required to be physically cross-tied for an operable 2EDG2 to serve as the 2VSF-9 emergency power source for Unit 1).

ATTACHMENT B

COMPONENT/TECH SPEC CROSS REFERENCE



INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONTIORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	$10^{-4} - 10^{1}$ R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	$1 - 10^{7}$ R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	5 & 6	$\leq$ 2 x background	$10 - 10^{6}$ cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	$\leq$ 2 x background	$10 - 10^{6}$ cpm	17, 20
c. Main Steam Line Radiation Monitors	1/Steam Line	1, 2, 3, & 4	Not Applicable	$10^{-1} - 10^{4}$ mR/hr	19

Note 1 - With fuel in the spent fuel pool or building.
Note 2 - MODES 1, 2, 3, 4, and during handling of irradiated fuel.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If performing CORE ALTERATIONS or moving irradiated fuel within the reactor building, secure the containment purge system or suspend CORE ALTERATIONS and movement of irradiated fuel within the reactor building.

b. If a containment PURGE is in progress, secure the containment purge system.

c. If continuously ventilating, verify the SPING monitor operable or perform the ACTIONS of 3.3.3.9, or secure the containment purge system.

ACTION 17 - With no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 18 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 20 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or within the next 6 hours initiate and maintain the control room emergency ventilation system in the recirculation mode of operation.

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR limit shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and at least one CEAC is operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the value specified in the CORE OPERATING LIMITS REPORT (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and at least one CEAC is operable); or

d. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

   1. Restore the DNBR to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the DNBR limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the DNBR to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

6) 3/4.2.1 - LINEAR HEAT RATE

With COLSS out of service, the linear heat rate shall be maintained ≤ 13.5 kW/ft.

7) 3.2.3 - AZIMUTHAL POWER TILT- T_q

The measured AZIMUTHAL POWER TILT shall be maintained ≤ 0.03.

8) 3/4.2.4 - DNBR MARGIN

The DNBR limit shall be maintained by one of the following methods:

a) With COLSS in service and neither CEAC operable - Maintain COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13%.

b) With COLSS out of service and at least one CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 4, using any operable CPC channel.

c) With COLSS out of service and neither CEAC operable - Operate within the Region of Acceptable Operation shown on Figure 5, using any operable CPC channel.

9) 3.2.7 - AXIAL SHAPE INDEX

The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a) COLSS IN SERVICE
- 0.27 ≤ ASI ≤ + 0.27

b) COLSS OUT OF SERVICE (CPC)
- 0.20 ≤ ASI ≤ + 0.20

FIGURE 4

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, CEAC OPERABLE)

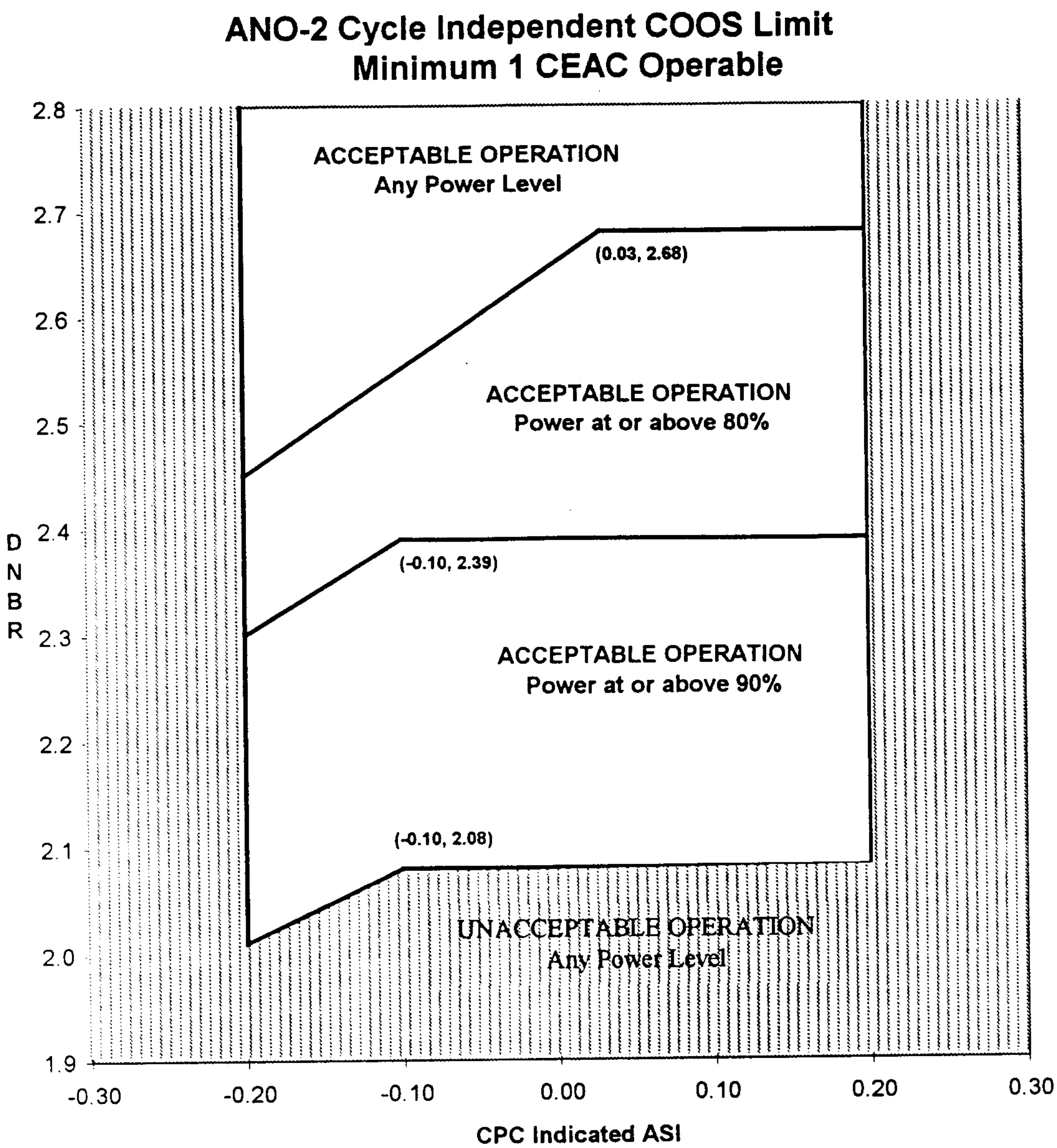
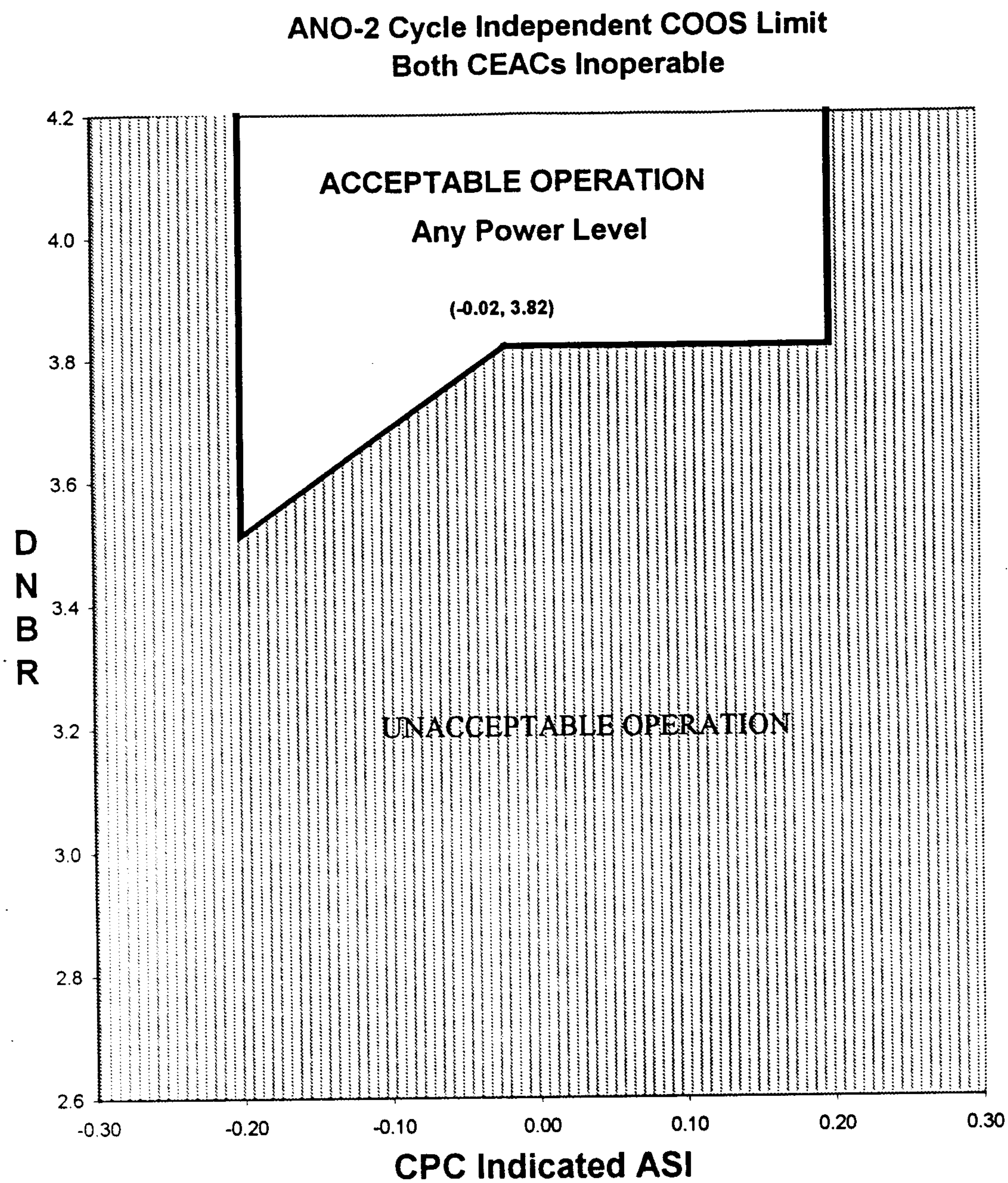


FIGURE 5

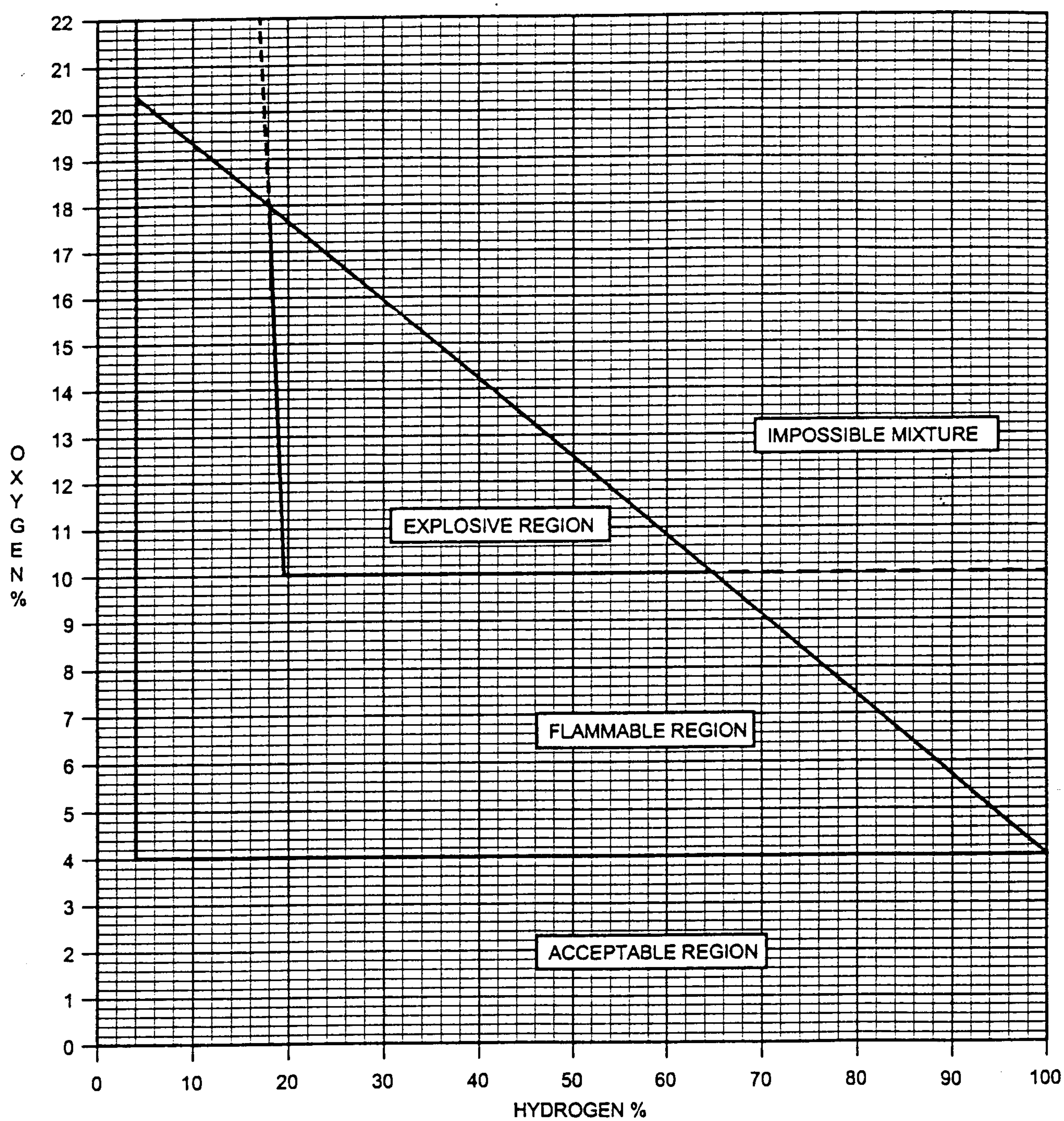
DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, BOTH CEACS INOPERABLE)



ATTACHMENT A

H_2/O_2 CONCENTRATIONS

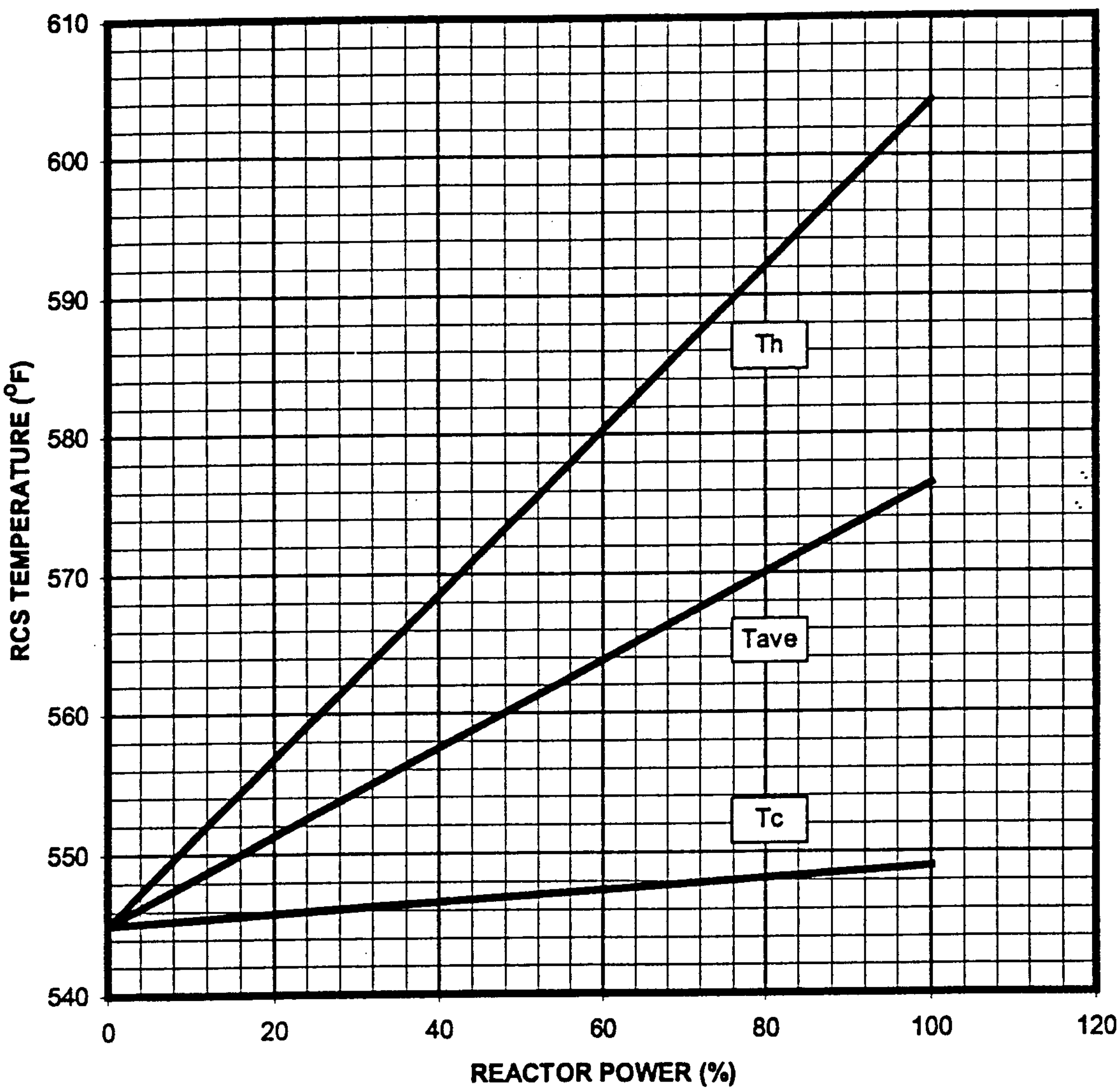


ATTACHMENT C

PAGE 1 OF 1

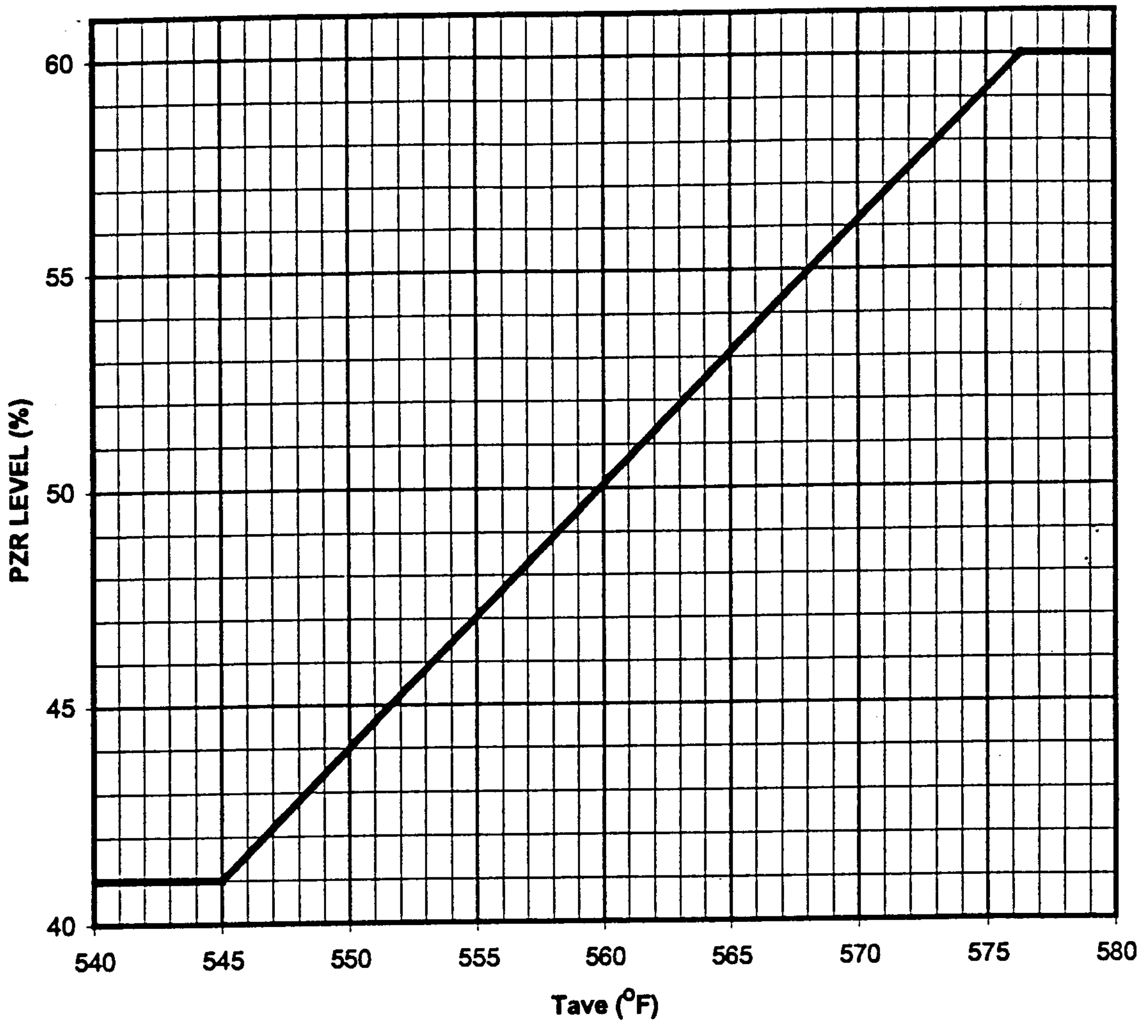
RCS TEMPERATURE VS REACTOR POWER

This temperature profile represents the desired trend for RCS temperature vs. Reactor power levels. The actual values at near full power may vary.



ATTACHMENT E

PRESSURIZER LEVEL PROGRAM



ATTACHMENT 1 - TEMPORARY ALTERATION SCREENING

Temporary alteration identification guidance should be incorporated into the Work Management Process. Expectations for Operations, Maintenance, Planners and System Engineering support for the Work Management Process include the initial identification of potential temporary alterations during the job planning stages. This screening criteria may be utilized during the workweek reviews.

The screening questions are:

1. Is the SSC affected by this activity [or associated SSC] in service, in standby or required to be operable?
2. Does the intended activity temporarily modify the SSC configuration so that it deviates from the approved drawings or other design documents or changes the design function of the SSC?
3. Would a design change or equivalency evaluation be required to make the altered configuration permanent?

If the answers to ALL three of these questions are "YES",

THE ACTIVITY IS A POTENTIAL TEMPORARY ALTERATION AND ENGINEERING ASSISTANCE MAY BE OBTAINED.

Proceed to TEMPORARY ALTERATION DETERMINATION GUIDELINES [ATTACHMENT 2]