Administrative Documents

(Yellow Paper)

SURRY EXAM
50-280, 50-281/2004-30

FEBRUARY 24 - MARCH 2
& MARCH 4, 2004 (WRITTEN)

1. Exam Preparation Checklist ES-201-1
2. Exam Outline Quality Checklist ES-201-2
3. Exam Security Agreement ES-201-3
4 Administrative Topics Outline (Final) ES-301-1
5 Control Room Systems and Facility Walk-through Test Outline (Final) ES-301-2
6 Operating Test Quality Check Sheet ES-301-3
7. Simulator Scenario Quality Check Sheet ES-301-4
8. Transient and Event Checklist ES-301-5
9. Competencies Checklist ES-301-6
10. Written Exam Quality Check Sheet .... ES-401-~~7~~ 6
11. ~~Written Exam Review Worksheet ....~~ ~~ES-401-9~~ N/A
FACILITY COMMENTS TO DRAFT WRITTEN EXAM
12. Written Exam Grading Quality Checklist ES-403-1
13 Post-Exam Check Sheet ES-501-1

Facility: SURRY Date of Examination: 2/23/04

Examinations Developed by: Facility / NRC (circle one)

Target Date*	Task Description / Reference	Chief Examiner's Initials
-180	1. Examination administration date confirmed (C.1.a; C.2.a & b)	8/28 SB
-120	2. NRC examiners and facility contact assigned (C.1.d; C.2.e)	10/27 SB
-120	3. Facility contact briefed on security & other requirements (C.2.c)	10/27 SB
-120	4. Corporate notification letter sent (C.2.d)	10/27 SB
[-90]	[5. Reference material due (C.1.e; C.3.c)]	11/26 SB
-75	6. Integrated examination outline(s) due (C.1.e & f; C.3.d)	12/11 SB
-70	7. Examination outline(s) reviewed by NRC and feedback provided to facility licensee (C.2.h; C.3.e)	12/16 SB
-45	8. Proposed examinations, supporting documentation, and reference materials due (C.1.e, f, g & h; C.3.d)	1/10 SB
-30	9. Preliminary license applications due (C.1.l; C.2.g; ES-202)	1/25 SB
-14	10. Final license applications due and assignment sheet prepared (C.1.l; C.2.g; ES-202)	2/9 SB
-14	11. Examination approved by NRC supervisor for facility licensee review (C.2.h; C.3.f)	2/9 SB
-14	12. Examinations reviewed with facility licensee (C.1.j; C.2.f & h; C.3.g)	2/9 SB
-7	13. Written examinations and operating tests approved by NRC supervisor (C.2.i; C.3.h)	2/16 SB
-7	14. Final applications reviewed; assignment sheet updated; waiver letters sent (C.2.g, ES-204)	2/16 SB
-7	15. Proctoring/written exam administration guidelines reviewed with facility licensee and authorization granted to give written exams (if applicable) (C.3.k)	2/16 SB
-7	16. Approved scenarios, job performance measures, and questions distributed to NRC examiners (C.3.i)	2/16 SB

* Target dates are keyed to the examination date identified in the corporate notification letter. They are for planning purposes and may be adjusted on a case-by-case basis in coordination with the facility licensee.

[ ] Applies only to examinations prepared by the NRC.

Facility: SURRY Date of Examination: 02-24-2004

Item	Task Description	Initials a	Initials b*	Initials c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model per ES-401.	MB	N/A	GL
	b. Assess whether the outline was systematically and randomly prepared in accordance with Section D.1 of ES-401 and whether all K/A categories are appropriately sampled.	MB	N/A	GL
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	MB	N/A	GL
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MB	N/A	GL
2. SIM	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, and major transients.	MB	N/A	GL
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity; ensure each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s)*, and scenarios will not be repeated on subsequent days.	MB	N/A	GL
	c. To the extent possible, assess whether the outline(s) conform(s) with the qualitative and quantitative criteria specified on Form ES-301-4 and described in Appendix D.	MB	N/A	GL
3. W/T	a. Verify that: (1) the outline(s) contain(s) the required number of control room and in-plant tasks, ✓ (2) no more than 30% of the test material is repeated from the last NRC examination, ✓ (3)* no tasks are duplicated from the applicants' audit test(s), and ✓ (4) no more than 80% of any operating test is taken directly from the licensee's exam banks. ✓	MB	N/A	GL
	b. Verify that: (1) the tasks are distributed among the safety function groupings as specified in ES-301, ✓ (2) one task is conducted in a low-power or shutdown condition. ✓ (3) 4 - 6 (2 - 3 for SRO-U) of the tasks require the applicant to implement an alternate path procedure, ✓ (4) one in-plant task tests the applicant's response to an emergency or abnormal condition, and ✓ (5) the in-plant walk-through requires the applicant to enter the RCA. 2 ✓	MB	N/A	GL
	c. Verify that the required administrative topics are covered.	MB	N/A	GL
	d. Determine if there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on subsequent days.	MB	N/A	GL
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam section.	MB	N/A	GL
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.	MB	N/A	GL
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	MB	N/A	GL
	d. Check for duplication and overlap among exam sections.	MB	N/A	GL
	e. Check the entire exam for balance of coverage.	MB	N/A	GL
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	MB	N/A	GL

	Printed Name / Signature	Date
a. Author	Mark A. Bates / [signature]	02-11-2004
b Facility Reviewer (*)	N/A	
c NRC Chief Examiner (#)	Steven D. Rose / [signature] Edwin Lea, Jr. / [signature]	2/11/04 2/11/04
d NRC Supervisor	Michael E. Ernstes / [signature]	2/18/04

Note: * Not applicable for NRC-developed examinations. ✓
Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

1. Pre-Examination

I acknowledge that I have acquired specialized knowledge about the NRC licensing examinations scheduled for the week(s) of 02-23-04 to 03-05-04 as of the date of my signature. I agree that I will not knowingly divulge any information about these examinations to any persons who have not been authorized by the NRC chief examiner. I understand that I am not to instruct, evaluate, or provide performance feedback to those applicants scheduled to be administered these licensing examinations from this date until completion of examination administration, except as specifically noted below and authorized by the NRC.Furthermore, I am aware of the physical security measures and requirements (as documented in the facility licensee's procedures) and understand that violation of the conditions of this agreement may result in cancellation of the examinations and/or an enforcement action against me or the facility licensee. I will immediately report to facility management or the NRC chief examiner any indications or suggestions that examination security may have been compromised.

2. Post-Examination

To the best of my knowledge, I did not divulge to any unauthorized persons any information concerning the NRC licensing examinations administered during the week(s) of 02-23-04 to 03-05-04. From the date that I entered into this security agreement until the completion of examination administration, I did not instruct, evaluate, or provide performance feedback to those applicants who were administered these licensing examinations, except as specifically noted below and authorized by the NRC.

	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATENOTE
1.	AMY C. EPPS	TRAINING ADMINISTRATOR	Amy C. Epps	11-03-03	Amy Epps	03-18-04
2.	JEFF T. SPENCE	SUPERVISOR - NT	[illegible]	1/9/04	[illegible]	3/8/04
3.	PHAT TRAN-LAM	SR SIM SUPPORT COORD	[illegible]	1/12/04	[illegible]	3/8/04
4.	Robert W Soderholm	Sr Sim Support Coord -NT	[illegible]	1/12/04	[illegible]	3/8/04
5.	AARON D BROWN	SR SIM SUPPORT COORD SOFTWARE	[illegible]	12Jan04	[illegible]	08Mar04
6.	Christopher G. Huth	Simulator Software	[illegible]	1-12-04	[illegible]	3/8/04
7.	WILLIAM W. MARSHALL	SR INST NUC OPS / LIC CLASS INST	[illegible]	1/13/04	[illegible]	3/8/04
8.	William B Gross	Shift Manager / Human Performance	[illegible]	1/20/04	[illegible]	3/16/04
9.	James R. Borden	Reactor Operator / ORB	[illegible]	01-21-04	[illegible]	3-24-04
10.	William J. Parker	Shift Manager	[illegible]	1-21-04	[illegible]	3-24-04
11.	Curtis L. Dardeau III	Admin - Ops Trng	[illegible]	2-09-04	[illegible]	3-24-04
12.	Carl F. Irwin III	NUC instr / Lic class instr	[illegible]	2-24-4	[illegible]	3/8/04
13.	David T. Llewellyn	Manager Nuclear Training	[illegible]	2/24/04	[illegible]	3/8/04
14.	J. Hal Warren JR	Sr Instructor - Nuc Ops	[illegible]	2/25/04	[illegible]	3/8/04
15.	JOHN HARTKA	UNIT SUPERVISOR	John Hartka	2-26-04	John Hartka	03-24-04

NOTES:

1. Pre-Examination

2-23-04 TO

I acknowledge that I have acquired specialized knowledge about the NRC licensing examinations scheduled for the week(s) of 3-05-04 as of the date of my signature. I agree that I will not knowingly divulge any information about these examinations to any persons who have not been authorized by the NRC chief examiner, I understand that I am not to instruct, evaluate, or provide performance feedback to those applicants scheduled to be administered these licensing examinations from this date until completion of examination administration, except as specifically noted below and authorized by the NRC. Furthermore, I am aware of the physical security measures and requirements (as documented in the facility licensee's procedures) and understand that violation of the conditions of this agreement may result in cancellation of the examinations and/or an enforcement action against me or the facility licensee. I will immediately report to facility management or the NRC chief examiner any indications or suggestions that examination security may have been compromised.

2. Post-Examination

To the best of my knowledge, I did not divulge to any unauthorized persons any information concerning the NRC licensing examinations administered during the week(s) of 02-23-04 to 03-05-04 From the date that I entered into this security agreement until the completion of examination administration, I did not instruct, evaluate, or provide performance feedback to those applicants who were administered these licensing examinations, except as specifically noted below and authorized by the NRC.

	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	GLENN D. JACKSON	INSTRUCTOR/OPS PROGRAM	Glenn D Jackson	11/19/03	Glenn D Jackson	3-9-04	
2.	—	CHANGE COORDINATOR	—	—			
3.	Terri Ware	Occ Health Nurse	Terri Ware RN	2/26/04	Terri Ware RN	3/23/04	
4.							
5.							
6.							
7.							
8.							
9.							
10.							
11.							
12.							
13.							

NOTES:

Facility: Surry **Examination Level (Underline one):** <u>RO</u> / SRO	**Date of Examination:** FEB2004 **Operating Test Number: 2004-301**

Administrative Topic (see Note)	Describe activity **to** be performed
Conduct of Operations	Calculate the Maximum Allowable Reactor Vessel Hydrogen **Venting** Time G2.1.23 (3.9/4.0); G2.1.25 (2.8/3.1)
Conduct of Operations	Shutdown Margin Calculation at Zero Power G2.1.7 (3.7/4.4)
Equipment Control	Construct Tagout for 1-RT-P-1C (SG Recirc & Transfer Pump) G2.2.13 (3.6/3.8)
Radiation *Control*	Dose / Stay Time Calculation G2.3.1 (2.6/3.1); **G2.3.4** (2.5/3.1)
Emergency Plan	N/A

NOTE: All items *(5* total) are required for SROs. RO applicants require only **4** items unless they are retaking only the administrative topics, when 5 are required.

Facility: Surry **Date of Examination:** FEB2004
Examination bevel (Underline one): RO / SRO **Operating Test Number:** 2004-301

Administrative Topic (see Note)	Describe activity to be performed
Conduct of Operations	Calculate the Maximum Allowable Reactor Vessel Hydrogen Venting Time G2.1.23 (3.9/4.0); G2.1.25 (2.8/3.1)
Conduct of Operations	Shutdown Margin Calculation at Zero Power G2.1.7 (3.7/4.4)
Equipment Control	Construct Tagout for 1-RT-P-1C (SG Recirc & Transfer Pump) G2.2.13 (3.6/3.8)
Radiation Control	**Dose** / Stay Time Calculation G2.3.1 (2.6/3.1); 62.3.4 (2.5/3.1)
Emergency Plan	Emergency Classification 2.4.41 (2.3/4.1); 2.4.44 (2.1/4.0)

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when 5 are required.

Facility: Surry **Date of Examination:** FEB2004
Exam Level (underline one): RO / SRO(I) / SRO(U) **Operating Pest No.:** 2004-301

Control Room Systems (8 for WO; 7 for SRO-I; 2 or 3 for SRO-U)

System / JPM Title	Type Code*	Safety Function
a. Start 2" WCP / High Vibration 015M1.23 (3.1/3.2)	N A S	4 primary
b. Place Hydrogen Analyzer In Service Following LOCA (58.01) 028A4.03 (3.1/3.3)	D S	5
C. 0-AP-22.00, Fuel Handling Abnormal Condition Immediate Actions (36.08) 034A2.01 (3.6/4.4); 036AK1.01 (3.5/4.1)	D A S	8
d. Restore Offsite Power to 1H 4160V Emergency Bus IAW AP-10.08 (18.06) 062A4.01 (3.3/3.1); 055EA2.06 (3.7/4.1)	D S	6
e. Transfer to Hot Leg Recirculation with 1 Charging Pump in Service (52.02) - ESF 011EAI.I 1(4.2/4.2)	M A S	3
f. Swap operating Main Feedwater Pumps due to problems with operating pump 059A4.08 (3.0/2.9)	N L S	4 secondary
g. Response to failed low Pressurizer bevel Channel (38.07) 028AA1.08 (3.7/3.6)	D S	2
lh. Remove a failed Source Range NI from Service During a Reactor Startup (62.02) 015A2.02 (3.1/3.5); 015A4.03(3.8/3.9)	D S	7
In-Plant Systems (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
i. Cross-Tie Unit 2 Emergency Ruses for Circulating Water Isolation (35.02) 062A2.12 (3.2/3.6); 062AA2.02 (2.9/3.6); 876A2.01 (3.5/3.7); 056AA1.02(4.0/3.9)	D	6
j. Locally Emergency Borate per AOP-3.0, Emergency Boration (41.01B) 024AA1.04 (3.6/3.7)	D A R	1

k. Cross-Connect Turbine Building Instrument Air (17.02) 065AK3.04 (3.0/3.2); 078K1.03 (3.3/3.4); 065AK3.08(3.7/3.9);069AA2.02(3.9/4.4)	D R	8

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

Facility: Surry	**Date of Examination:** FEB2004
Exam Level (underline one): RO / SRO(I) / SRO(U)	**Operating Test No.:** 2004-301

Control Room Systems (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)

System / JPM Title	Type Code*	Safety Function
a. Start 2nd RCP 1High Vibration 015AA1.23 (3.1/3.2)	N A S	$4_{primary}$
b.		
C. 0-AP-22.00, Fuel Handling abnormal Condition Immediate Actions (36.06) 034A2.01 (3.6/4.4); 036AK1.0t (3.5/4.1)	D A S	8
d. Restore Offsite Power to 1H 4160V Emergency Bus IAW AQ-10.08 (18.06) 06284.01 (3.313.1); 055EA2.06 (3.7/4.1)	D S	6
e. Transfer to Hot Leg Recirculation with I Charging Pump in Service (52.02) - ESF 011EA1.11 (4.2/4.2)	M A S	3
f. Swap operating Main Feedwater Pumps due to problems with operating pump 059A4.08 (3.0/2.9)	N L S	$4_{secondary}$
g. Response to failed low Pressurizer Level Channel (38.07) 028AA1.08 (3.7/3.6)	D A S	2
h. Remove a failed Source Range NI from Service During a Reactor Startup (62.02) 015A2.02 (3.1/3.5); 015A4.03 (3.8/3.9)	D S	7

In-Plant Systems (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)

i. Cross-Tie Unit 2 Emergency Buses for Circulating Wafer Isolation (35.02) 062A2.12 (3.2/3.6); 062AA2.02 (2.9/3.6); 076A2.01 (3.5/3.7); 056AA1.02 (4.0/3.9)	D	6
j. Locally Emergency Borate per AOP-3.0, Emergency Boration (41.01B) 024AA1.04 (3.6/3.7)	D A R	1

k. Cross-Connect Turbine Building Instrument Air (17.02) 065AK3.04 (3.0/3.2); 078K1.03 (3.3/3.4); 065AK3.08(3.7/3.9);069AA2.02(3.9/4.4)	D R	8

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

Facility: Surry	**Date of Examination:** FEB2004
Exam Level (underline one): RO / SRO(I) / SRO(U)	**Operating Test No.:** 2004-301

Control Room Systems (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)		
System / JPM Title	Type Code*	Safety Function
El. Start 2" RCP / High Vibration 015881.23 (3.1/3.2)	N A S	$4_{primary}$
b.		
c.		
d.		
e. Transfer to Hot Leg Recirculation with 1 Charging Pump in Service (52.02) - ESF 011EA1.11 (4.2/4.2)	M A S	3
f. Swap operating Main Feedwater Pumps due to problems with operating pump 053A4.08 (3.0/2.9)	N L S	$4_{secondary}$
g.		
h.		
In-Plant Systems (3 for RO; 3 tor SRO-I; 3 or 2 for SRO-U)		
i.		
j. Locally Emergency Borate per AOP-3.0, Emergency Boration (41.01B) 024AA1.04 (3.6/3.7)	D A R	1

k. Crass-Connect Turbine Building Instrument Air (17.02) 065AK3.04 (3 0/3.2); 078K1.03(3.3/3.4); 065AK3.08(3.7/3.9);069AA2.02(3.9/4.4)	D R	8
* **Type Codes: (D)irect from bank, (M)odified** from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, &)ow-Power, (R)CA		

Facility: SURRY Date of Examination 02-24-2004 Operating Test Number: 2004-301

1. GENERAL CRITERIA		Initials a	b*	c#
a.	Tho operating test conforms wil the previously approved outline; changes a with requ rements 10 CFR 55.45 operational importance safety distribution).	MB	N/A	SB
b.	There is no day-to-day repetition between this and other operating tests to be administered during this examination.	MB	N/A	SB
c.	The operating test shall not duplicate items from the applicants' audit test(s)(see Section D.1.a). NRC Developed Exam Independent of Audit	MB	N/A	SB
d.	Overlap with the written examination and between different parts of the operating test is within acceptable limits.	MB	N/A	SB
e.	It appears that the operating test will differentiate between competent and less-than-competent applicants at the designated license level.	MB	N/A	SB
2. WALK-THROUGH		--	--	--
a.	Each JPM includes the following, as applicable: . initial conditions ✓ . initiating cues ✓ · references and tools, including associated procedures ✓ . reasonable and validated time limits (average time allowed for completion) and specific designation if deemed to be time critical by the facility licensee ✓ . specific performance criteria that include: - detailed expected actions with exact criteria and nomenclature ✓ - system response and other examiner cues ✓ - statements describing important observations to be made by the applicant ✓ - criteria for successful completion of the task ✓ - identification of critical steps and their associated performance standards ✓ - restrictions on the sequence of steps, if applicable ✓	MB	N/A	SB
b.	Repetition from operating tests used during the previous licensing examination is within acceptable limits (30% for the walk-through) and do not compromise test integrity.	MB	N/A	SB
c.	At least 20 percent of the JPMs on each test are new or significantly modified.	MB	N/A	SB
3. SIMULATOR CRITERIA		--	--	--
a.	The associated simulator operating tests (scenario sets) have been reviewed in accordance with Form ES-301-4 and a copy is attached.	MB	N/A	SB

	Printed Name / Signature	Rate
a. Author	MARK A. BATES / [signature]	02-11-2004
b. Facility Reviewer(*)	N/A	
c. NRC Chief Examiner (#)	STEVEN D. ROSE / [signature] Edward Lea, Jr / [signature]	2/12/04 2/17/04
d. NRC Supervisor	MICHAEL E. ERNSTES / [signature]	2/18/04

NOTE: * The facility signature is not applicable for NRC-developed tests.
Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

Facility: SURRY Date of Exam 02-24-2004 Scenario Numbers 1 / 2 / 3 +spare Operating Test No.: 2004-301

QUALITATIVE ATTRIBUTES		Initials a	Initials b*	Initials c#
1.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue the operators into expected events.	MB	N/A	[illegible]
2.	The scenarios consist mostly of related events.	MB	N/A	[illegible]
3.	Each event description consists of · the point in the scenario when it is to be initiated · the malfunction(s) that are entered to initiate the event · the symptoms/cues that will be visible to the crew · the expected operator actions (by shift position) · the event termination point (if applicable)	MB	N/A	[illegible]
4.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.	MB	N/A	[illegible]
5.	The events are valid with regard to physics and thermodynamics.	MB	N/A	[illegible]
6.	Seq [illegible] and [illegible] of events is [illegible] and allows the examination team to obtsin complete evaluation [illegible] the scenario objectives.	MB	N/A	[illegible]
7.	If time compression techniques are used, the scenario summary clearly so indicates. Operators have sufficient time to carry out expected activities without undue time ccnstmints. Cues are given.	MB	N/A	[illegible]
a.	The simulator modeling is not altered.	MB	N/A	[illegible]
9.	The scenarios have been validated. Pursuant to 10 CFR 55.46(d), any open Simulator performance deficiencies have been evaluated to ensure that functional fidelity is maintained while running the planned scenarios.	MB	N/A	[illegible]
10.	Every operator will be evaluated using at least one new or significantly modified scenario. All other scenarios have been altered in accordance with Sectbn D.5 of ES-301.	MB	N/A	[illegible]
11.	Ail individual operator competencies can be evaluated, as verified using Form ES-301-6 (submit the form along with the simulator scenarios).	MB	N/A	[illegible]
12.	Each applicant will be significantly involved in the minimum number of transients and events specified on Form ES-301-5 (submit the form with the simulator scenarios).	MB	N/A	[illegible]
13.	The level of difficulty is appropriate to support licensing decisions for each crew position.	MB	N/A	[illegible]

TARGET QUANTITATIVE ATTRIBUTES (PER SCENARIO; SEE SECTION D.5.d)		Actual Attributes	--	--	--
1.	Total malfunctions (5-8)	6 / 9 / 8	MB	N/A	[illegible]
2.	Malfunctions after EOP entry (1-2)	1 / 2 / 3	MB	N/A	[illegible]
3.	Abnormal events (2-4)	2 / 0 / 1	MB	N/A	[illegible]
4.	Major transients (1-2)	1 / 1 / 1	MB	N/A	[illegible]
5.	EOPs entered/requiring substantive actions (1-2)	1 / 1 / 1	MB	N/A	[illegible]
6.	EOP contingencies requiring substantive actions (0-2)	0 / 1 / 1	MB	N/A	[illegible]
7.	Critical tasks (2-3)	2 / 1 / 2	MB	N/A	[illegible]

OPERATING **TEST** NO.: I1

Applicant Type	Evolution Type	Minimum Number	Scenario Number 1 RO	1 BOP	2 RO	2 BOP	3 RO	3 BOP	4 RO	4 BOP
RO	Reactivity	1*								
	Normal	1*								
	Instrument / Component	4*								
	Major	1								
SRO-I As RO	Reactivity	1*					1			
	Normal	0								
	Instrument / Component	2*					2 6			
	Major	1					7			
SRO-I As SRO	Reactivity	0								
	Normal	1*		2						
	Instrument / Component	2*	1,3 5	4 6						
	Major	1	7							
SRO-U	Reactivity	0								
	Normal	1*								
	Instrument / Component	2*								
	Major	1								

Instructions: (1) E tl ai test e and I 1 ent n for each evolution t

(2) y mani may be conducted under l or cc abnormal itions (r e Se i) t must be significant per Sec i C.2.a f Appendix D. * Reactivi and n r l ol iti may be ol with litio l in ument or t malfunctions on a one-for-one basis.

W e er pra a, both in and component malfunctio should be included; only th at require verifiable that id insight to i competence count toward the minimum requirement

Author: MARK A. BATES [signature] 02.12.04

NRC Reviewer: STEVEN D. ROSE [signature] 2/11/04

OPERATING TEST **NO.:** 12

Applicant Type	Evolution Type	Minimum Number	Scenario Number							
			1		2		3		4	
			RO	BOP	RO	BOP	RO	BOP	RO	BOP
RO	Reactivity	1*								
	Normal	1*								
	Instrument / Component	4*								
	Major	1								
SRO-I As RO	Reactivity	1*								
	Normal	0								
	Instrument / Component	2*	1,3 5							
	Major	1	7							
SRO-I As SRO	Reactivity	0					1			
	Normal	1*								
	Instrument / Component	2*					2 6	3 4 5		
	Major	1					7			
SRO-U	Reactivity	0								
	Normal	1*								
	Instrument / Component	2*								
	Major	1								

Instructions:

(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. * Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.

(3) Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirement.

Author: MARK A. BATES [signature] 02-12-04

NRC Reviewer: STEVEN D. ROSE [signature] 2/13/04

OPERATING TEST NO.: 13

Applicant Type		Evolution Type	Minimum Number	Scenario 1 RO	Scenario 1 BOP	Scenario 2 RO	Scenario 2 BOP	Scenario 3 RO	Scenario 3 BOP	Scenario 4 RO	Scenario 4 BOP
RO		Reactivity	1*								
		Normal	1*								
		Instrument / Component	4*								
		Major	1								
SRO-I	As RO	Reactivity	1*					1			
		Normal	0								
		Instrument / Component	2*					2 ~~3~~ 6			
		Major	1					7			
	As SRO	Reactivity	0								
		Normal	1*								
		Instrument / Component	2*			2 3 4	1 6 7				
		Major	1			8					
SRO-U		Reactivity	0								
		Normal	1*								
		Instrument / Component	2*								
		Major	1								

Instructions:

(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. * Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.

(3) Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirement.

Author: Mark A. Bates [signature] 02-12-04

NRC Reviewer: Steven D Rose [signature] 2/17/04

OPERATING TEST NO.: Crew 1/2 ($U_{1/2}$, $R_{1/3}$, $R_{2/4}$)

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1 RO	1 BOP	2 RO	2 BOP	3 RO	3 BOP	4 RO	4 BOP
RO		Reactivity	1*								
		Normal	1*		2						
		Instrument / Component	4*	1 3 5	4 6	2 3 4	1 6 7				
		Major	1	7		8	8				
SRO-I	As RO	Reactivity	1*								
		Normal	0								
		Instrument / Component	2*								
		Major	1								
	As SRO	Reactivity	0								
		Normal	1*								
		Instrument / Component	2*								
		Major	1								
SRO-U		Reactivity	0								
		Normal	1*		2						
		Instrument / Component	2*	1 3 5	4 6	2 3 4	1 6 7				
		Major	1	7		8					

Instructions:

(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must **be** significant per Section C.2.a of Appendix D. * Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions **on a** one-for-one basis.

(3) Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirement.

Author: MARK A. BATES [signature] 02-12-04

NRC Reviewer: STEVEN D. ROSE [signature] 2/12/04

OPERATING TEST NO.: I4

Applicant Type	Evolution Type	Minimum Number	Scenario Number 1 RO	1 BOP	2 RO	2 BOP	3 RO	3 BOP	4 RO	4 BOP
RO	Reactivity	1*								
	Normal	1*								
	Instrument / Component	4*								
	Major	1								
SRO-I As RO	Reactivity	1*								
	Normal	0								
	Instrument / Component	2*			2 3 4					
	Major	1			8					
SRO-I As SRO	Reactivity	0					1			
	Normal	1*								
	Instrument / Component	2*					2 6	3 4 5		
	Major	1					7			
SRO-U	Reactivity	0								
	Normal	1*								
	Instrument / Component	2*								
	Major	1								

Instructions:

(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix **D.** * Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.

(3) Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirement.

Author: MARK A. BATES [signature] 02-11-04

NRC Reviewer: STEVEN D ROSE [signature] 2/11/04

Competencies	SRO				RO				BOP			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3	spare 4	1	2	3	spare 4	1	2	3	spare 4
Interpret/ Diagnose Events and Conditions	1, 3,4,5,6 ,7,8	1,2, 3.4. 5.6. 7.8. 9,10	2,3,4, 5,6,7, 8,9,10	2,3,5, 6,7,8, 9,10, 11,12	1,3, 5,7	2,3, 4,5, 8, 10	2,3, 7,8, 9	2,3,8 ,12	4,6, 7,8	1,6, 7,8, 9	4,5, 6,7, 10	5,6,7, 9,10, 11,12
Comply With and Use Procedures (1)	1,2,3,4 5,6,7, 8	1.2.3.4 ,5,6,7, 8,9,10	1,2,3, 4,5,6, 7,8,9, 10	1,2,3, 4,5,6, 7,8,9, 10,11, 12	1,3, 5,7	2,3, 4,5, 8, 10	1,2, 3,7, 8, 9	1,2,3 ,4,8, 12	2,4, 6,7, 8	1,6, 7,8, 9	4,5, 6,7, 10	4,5,6, 7,9, 10,11 ,12
Operate Control Boards (2)	N/A	N/A	N/A	N/A	1,3, 5,7	2,3, 4,5, 8, 10	1,2, 3,7, 8, 9	2,3,8 ,12	2,4, 6,7, 8	1,6, 7,8, 9	4,5, 6,7, 10	5,6,7, 9,10, 11,12
Communicate and Interact	1,2,3,4 ,5,6,7, 8	1,2,3,4 ,5,6,7, 8,9,10	1,2,3, 4,5,6, 7,8,9, 10	1,2,3, 4,5,6, 7,8,9, 10,11, 12	1,3, 5,7	2,3, 4,5, 8, 10	1,2, 3,7, 8, 9	1,2,3 ,4,8, 12	2,4, 6,7, 8	1,6, 7,8, 9	4,5, 6,7, 10	4,5,6, 7,9, 10,11 ,12
Demonstrate Supervisory Ability (3)	1,2,3,4 ,5,6,7, 8	1,2,3,4 ,5,6,7, 8,9,10	1,2,3, 4,5,6, 7,8,9, 10	1,2,3, 4,5,6, 7,8,9, 10,11, 12	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Comply With and Use Tech. Specs. (3)	4 5	3 4	4 6	2,3, 5	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Notes:

(1) includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.

Instructions:
Circle the applicant's license type and enter one or more event numbers that will allow the examiners to evaluate every applicable competency for every applicant.

Author: MARK A. BATES [signature] 02-12-04

NRC Reviewer: STEVEN D. ROSE [signature] 2/17/04

Facility: SURRY Date of Exam: 02-24-2004 Exam Level: RO/SRO

	Item Description	Initial a	Initial b*	Initial c#
1.	Questions and answers technically accurate and applicable to facility	MB	N/A	SR
2.	a. NRC K/As referenced for all questions b. Facility learning objectives referenced as available	MB	N/A	SR
3.	SRO questions are appropriate per Section D.2.d of ES-401	MB	N/A	SR
4.	Question selection and duplication from the last two NRC licensing exams appears consistent with a systematic sampling process			SR
5.	Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: __ the audit exam was systematically and randomly developed; or __ the audit exam was completed before the license exam was started; or ✓ the examinations were developed independently; or __ the licensee certifies that there is no duplication; or __ other (explain)	MB	N/A	SR
6.	Bank use meets limits (no more than 75 percent from the bank at least 10 percent new, and the rest modified); enter the actual RO / SRO-only question distribution(s) at right — Bank: 29 / 8; Modified: 7 / 2; New: 39 / 15	MB	N/A	SR
7.	Between 50 and 60 percent of the questions on the RO exam are written at the comprehension/analysis level; the SRO exam may exceed 60 percent if the randomly selected K/As support the higher cognitive levels; enter the actual RO / SRO question distribution(s) at right — Memory: 31 / 9; C/A: 44 / 16	MB	N/A	SR
8.	References/handouts provided do not give away answers	MB	N/A	SR
9.	Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the Tier to which they are assigned; deviations are justified	MB	N/A	SR
10.	Question psychometric quality and format meet ES, Appendix E, guidelines	MB	N/A	SR
11.	The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with value on cover sheet	MB	N/A	SR

	Printed Name / Signature	Date
a. Author	MARK A. BATES / Mark A. Bates	02-11-2004
b. Facility Reviewer (*)	N/A	
c. NRC Chief Examiner (#)	STEVEN D. ROSE / S.D. Rose / Edwin Lea, Jr. / Edwin Lea, Jr.	2/11/04
d. NRC Regional Supervisor	MICHAEL E. ERNSTES / Michael E. Ernstes	2/11/04

Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations.
Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

FACILITY Comments

to the

Draft Written Exam

Surry 2004-301

RO & SRO

1. 003K4.03 1

Which ONE of the following correctly describes the Reactor Coolant Pump (RCP) bearing oil lift system?

A. The oil lift pump discharge pressure must be greater than 350 psig prior to RCP start. Once the RCP reaches operating speed. the thrust runner circulates oil in the upper and lower bearing assemblies.

B. The oil lift pump discharge pressure must be greater than 300 psig prior to RCP start. Once the RCP reaches operating speed, the RCP Oil Lift System supplies the bearing lubrication.

C. The oil lift pump discharge pressure must be greater than 350 psig prior to RCP start. Once the RCP reaches operating speed, the RCP Oil Lift System supplies the bearing lubrication.

D. The oil lift pump discharge pressure must be greater than 300 psig prior to RCP start. Once the RCP reaches operating speed, the thrust runner circulates oil in the upper and lower bearing assemblies.

Surry

References:
ND-88.1-LP-6, Reactor Coolant Pumps, Rev. 16

Distractor Analysis:
A. Correct because there is a 350 psig discharge interlock with respective RCP. The Oil Lift Pump ensures adequate lubrication upon RCP start, but once the pump reaches operating speed, the thrust runner acts as an oil pump and circulates oil in the upper and lower bearing assemblies.
B. Incorrect because pressure interlock is at 350 psig, not 300 psig.
C. Incorrect because thrust runner circulates oil in upper and lower reservoir, not the Oil Lift System.
D. Incorrect because pressure interlock is at 350 psig, not 300 psig.

003 Reactor Coolant Pumps
K4.03: Knowledge of RCPs design feature(s) and / or interlock(s) which provide for the folfowing: Adequate lubrication of the RCP.
Answer: A

2. 004A2.17 1

The following Unit 1 conditions exist:
- Operating at 85% power
- Pressurizer pressure control is in its normal configuration
- A Pressurizer Safety Valve is leaking
- 1C-B8, PRZR LO PRESS, annunciates
- 1-AP-34.00, Increasing or Decreasing RCS Pressure, has been entered

Which ONE of the following correctly describes the affect on charging flow and an appropriate mitigating action in accordance with 1-AP-31.00?

A. Charging flow initially increases. Place the PRZR PRESS MASTER CNTRL in MANUAL and increase the demand to try to stop the pressure decrease.

B. Charging flow initially decreases. Place the PRZR PKESS MASTER CNTRL in MANUAL and increase the demand to try to stop the pressure decrease.

C. Charging flow initially increases. Place the PRZR PRESS MASTER CNTRL in MANUAL and decrease the demand to try to stop the pressure decrease.

5. Charging flow initially decreases. Place the PRZR PRESS MASTER CNTRL in MANUAL and decrease the demand to try to stop the pressure decrease.

References:
NB-93.3-LP-5, Pressurizer Pressure Control, Rev. 9
ND-88.3-LP-2, Charging and Letdown, Rev. 10
1-AP-31.00, Increasing or Decreasing RCS Pressure, Rev. 4

Bistractor Analysis:
A. Incorrect because increasing the demand will lower pressure, not increase it.
B. Incorrect because charging Row will not initially decrease and increasing the demand will lower pressure, not increase it.
C. Correct because charging flow will initially increase due to the sudden pressure drop in the RCS. Also, decreasing the demand on the controller while in manual will act to try to raise pressure.
5. Incorrect because charging flow will not initially decrease.

004 Chemical and Volume Control
A2.17: Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) **based** on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low PZR pressure.
Answer: C

3. 005K5.02 1

The following Unit 1 conditions exist:
- RCS level is 12.5 feet on 1-RC-LI-100A
- RCS level is 12 feet 5 inches on 1-RC-LR-105
- A loss of decay heat removal has occurred and 1-AP-27.00, Loss of Decay Heat Removal Capability, has been entered.
- The RHR system has just been made available.

Which ONE of the following methods per 1-AP-27.00 should be used to sweep air from the RHR lines during a loss of decay heat removal capability if inadequate time exists to completely vent the RHR System prior to boiling in the core?

A. Refill the RCS to 13.5 feet, verify 10 ^{o}F subcooling, and run an RHR pump at a flow rate of > 2950 gpm.

B. Maintain RCS level at 12.5 feet, verify subcooling, and run an RHR pump at a flow of > 2950 gpm.

C. Maintain RCS level at 12.5 feet, verify subcooling, and run an RHR pump at a flow of < 2950 gpm.

D. Close RH-MOV-1720A and B, RHR Outlets, then open "A" and "C" Safety Injection Accumulator Isolation MOVs.

References:
NB-88.2-LP-1, Residual Heat Removal System Description, Rev. 8
ND-88.2-LP-2, Operation of Residual Heat Removal System, Rev. 15
NB-88.2-LP-3, Braindown and Midloop Operations, Rev. 12
1-AP-27.00, Loss of Decay Heat Removal Capability, Rev. 10

Distractor Analysis:
A. Correct because based on procedural Note in 1-AP-27.00, Page 16 ~~of~~ of19, Rev. 15.
B. Incorrect because RCS needs to be filled to 13.5 feet.
C. Incorrect because RCS needs to be filled to 13.5 feet. Also flow needs to be greater than 2950 gpm.
B. Incorrect because no procedural guidance exists to support the actions.

Surry ILT Exam Bank Question #275

005 Residual Heat Removal
K5.02: Knowledge of the operational implications of the following concepts as they apply to the WHRS: Need for adequate subcooling.
Answer: **A**

4. 006K6.03 1

Unit 1 tripped 30 minutes ago. The following plant conditions currently exist:

- 7 CETCs indicate between 700 °F and 750 °F and slowly rising
- No RCPs are operating
- RVLIS Full Range is indicating 46% and slowly lowering
- Steam Generator levels are 20% and rising
- Subcooling based on CETCs is 0 °F
- E-0, Reactor Trip or Safety Injection, has been exited and Safety Function Status Trees are being monitored
- FR-C.1, Response to Inadequate Core Cooling, has been implemented
- RCP Seal Injection flow is 3 gprn to all RCPs
- RCP Seal delta-Ps are all approximately 200 psid
- Source Range Startup Rate is zero
- Attempts to establish HHSI flow have failed

Which ONE of the following describes the correct strategy for mitigating the consequences of these conditions?

A. Depressurize all intact steam generators at maximum rate, but maintain steam flow less than 1.0 x 10^5 PPH to try establish accumulator and ~~RHR~~ LHSI flow. If CETCs rise above 1209 °F, then check conditions for starting a RCP.

B. Depressurize all intact steam generators at maximum rate, but maintain steam flow less than 1.0 x 10^5 PPH to try to establish accumulator and ~~RHR~~ LHSI flow. Even if CETC temperatures rise above 1200 °F, RCPs should not be started due to low RCP seal injection flow.

C. Depressurize all intact steam generators at maximum rate, but maintain steam flow less than 1.0 x 10^6 PPH to try to establish accumulator and ~~RHR~~ LHSI flow. Even if CETC temperatures rise above 1200 °F, RCPs should not be started due to low RCP seal injection flow.

D. Depressurize all intact steam generators at maximum rate, but maintain steam flow less than 1.0 x 10^6 PPH to try to establish accumulator and ~~RHR~~ LHSI flow. If CETCs rise above 1200 °F, then check conditions for starting a WCP.

Surry

References:
ND-95.3-LP-38, FR-C.I Response to Inadequate Core Cooling, Rev. 8
FR-C.1, Response to Inadequate Core Cooling, Rev. 18

Distractor Analysis:
A. Incorrect because 1.0 x 10^5 PPH is well below the MSIV closure setpoint and does not even approach the maximum rate (an entire order of magnitude low).
B. Incorrect because **4.0** x 10^5 PPH is well below the MSIV closure setpoint and does not even approach the maximum rate (an entire order of magnitude low).
C. Incorrect because RCPs should be started even when normal conditions not met.
D. Correct because procedural guidance exists to support the actions. MSIV closure will occur if flow *is* greater than 1.0 x 10^6 PPH. The purpose for the actions **is** to establish low head flow from accumulators and LHSI. RCP support criteria **is** desirable, but not a prerequisite for starting RCPs.

006 Emergency Core Cooling
K.6.03: Knowledge of the effect of a loss or malfunction on the following will have on ECCS: Safety Injection **Pumps.**
Answer: D

5. 007EK2.02 1

The following conditions exist:
- Unit 1 is at *90%* power
- Reactor protection testing is in progress
- Reactor Trip Breaker "**A**" is closed
- Reactor Trip Breaker "**B**" is open
- Reactor Trip Bypass Breaker "**B**" is racked in and closed

Which ONE of the following describes the plant response if reactor trip bypass breaker "A" is racked in and closed?

A. Both reactor trip bypass breakers "A" and "B" and reactor trip breaker "A" will trip open and the reactor will trip.

B. Only reactor trip bypass breakers "**A**" end "**B**" will trip open and the reactor will trip.

C. Reactor trip breaker "A" will trip open and the plant will remain at 90% power.

D. Reactor trip bypass breaker "A" will trip open and the plant will remain at 90% power.

Surry

References:
ND-93.3-LP-17, **AMSAC,** Rev. 10
ND-93.3-LP-10, Reactor Protection - General, Rev. 5

Distractor Anaysis:
A. Incorrect because reactor trip breaker "**A**" will not open.
B. Correct because this is the correct response per ND-93.3-LP-10.
C. Incorrect because reactor trip breaker "A" will not open and plant will trip.
D. Incorrect because the plant will trip.

Surry ILT Bank Question #1667

007 Reactor Trip Stabilization
EK2.02: Knowledge of the interrelationships between a reactor trip and the following: Breakers, relays, and disconnects.
Answer: B

6. 007K5.02 1

Given the following Unit 4 conditions:
- A heatup is in progress to return to power from a cold shutdown condition
- RCS is filled and vented
- Pressurizer is solid
- A nitrogen blanket has been established on the PWT
- PRT Level = 95%
- Pressurizer Heaters are energized

Which ONE of the following must be accomplished prior to drawing a bubble in the Pressurizer?

A. Drain the PRT to 60 - 80%.

B. Verify VCT oxygen concentration less than 3%.

C. Drain the Pressurizer to 22.2%.

D. Pressurize the RCS to 200 - 270 psig on PI-1-403, Nar Range.

Surry

References:
1-GOP-1.1, Unit Startup. RCS Heatup from Ambient to 495 Degrees F,, Rev. 25
1-OP-RC-011, Pressurizer Relief Tank Operations, Rev. 13

Distractor Analysis:
A. Correct because GOP-1.1 Step 5.5.4 directs establishment of normal PRT level prior to drawing a bubble. OP-RC-011 Step 5.1.1 states the normal PRT level to be 60 - 80%.
B. Incorrect because GOP-1.1 Step 5.5.6 requirement is to verify VCT oxygen **< 2%.**
C. Incorrect because this is an action following establishment of drawing **a** bubble (GOP-1.1, Step 5.5.13).
D. Incorrect because RCS should be between 300 and 370 psig on PI-1-403.

007 Pressurizer Relief / Quench Tank
K5.02: Knowledge of the operational implications of the following concepts **as** they apply to PRTS: Method of forming a steam bubble in the PZR.
Answer: **A**

7. 008AA2.06 1

Given the following Unit **1** conditions:
- Reactor power = 100%
- All other parameters are at normal steady state values
- Subsequently PT-444 fails high

Assuming no operator action **is** taken, which ONE of the following is correct?

A. PORV-1455C opens, pressure decreases to 2000 psig, PORV-1455C closes, and pressure stabilizes around 2000 psig.

B. PORV-1456 opens, pressure decreases lo 2000 psig, PORV-1456 closes, and pressure stabilizes around 2000 psig.

C. PORV-1455C opens, at 2000 psig PORV-1455C closes: however. pressure will continue to decrease causing a reactor trip and safety injection.

D. PQRV-1456 opens, at 2000 psig PORV-1456 closes: however, pressure will continue to decrease causing a reactor trip and safety injection.

Sur9

References:
ND-93.3-LP-5, Pressurizer Pressure Control, Rev. 9

Distractor Analysis:
A. Incorrect because both spray valves also open, which causes pressure to continue to decrease.
B. Incorrect because buth spray valves open, which causes pressure to continue to decrease Also incorrect because PORV-1456 does not open.
C. Correct because both spray valves open causing a reactor trip on OT-delta-T or Low Pressurizer Pressure, followed by SI.
D. Incorrect because PORV-1456 does not open.

008 Pressurizer Pressure Control
AA2.03: Ability to determine and interpret the following as they apply to the pressurizer vapor space accident: PQRV logic control under low-pressure conditions.
Answer: C

8. 008K1.02 1

Which ONE of the following correctly describes loads cooled by the Component Cooling Water (CCW) System or subsystem of CCW?

A. RCP bearing lube oil coolers, neutron shield tank coolers, RCP seal water return cooler, outside recirc spray pump seals.

B. HHSI pump seals, LHSI pump seals, RHR pump seals, RCP motor air coolers.

C. RHR pump seals, RCP bearing lube oil coolers, neutron shield tank coolers, HHSI pump seals

D. LHSI pump seals, RHW pump seals, RCP motor air coolers, neutron shield tank coolers.

Surry

Reference:
ND-88.5-LP-7, Component Cooling, **Rev. 19**
ND-88.3-LP-5, Charging System, Rev. 16

Distractor Analysis:
A. Incorrect because outside recirc spray pump seals are not cooled by CC.
B. incorrect because LHSI pump seals are not cooled by CC.
C. Correct because all are cooled by CC or a subsystem.
D. Incorrect because LHSI pump seals are not cooled by CC.

Requal Bank Question #527

008 Component Cooling
K1.02: Knowledge of the physical connections and / or cause-effect relationships between the CCWS and the following systems: Loads cooled by CCWS.
Answer: C

9. 008K4.01 1

Ten seconds after a Safety Injection occurs, the " A Component Cooling Pump trips. Which **ONE** of the following describes the operation of the CC pumps?

A. The "B" CC pump will not auto start without a required operator action.

B. The "B" CC pump will auto start 60 seconds after the "A" CC pump trips

C. The "B" CC pump will auto start as soon as the "A" CC pump trips

D. The "B" CC pump will auto start 50 seconds after the " A CC pump trips.

Surry

References:
ND-88.5-LP-1, Component Cooling Water System, Rev. 19.

Distractor Analysis:
A. Correct because Auto Start Inhibit due to SI will prevent auto start of the CC pump, but the pump may be manually started et any time.
B. *Incorrect* because the Auto Start Inhibit will block the auto start.
C. Incorrect because the Auto Start Inhibit will block the auto start.
D. Incorrect because the Auto Start Inhibit will block the auto start.

ILT Bank Question # 537

008 Component Cooling Water System
K4.01: Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump.
Answer: **A**

10. 010A1.01 1
The following plant conditions exist:
- Dilution to criticality has just been completed
- Operators note that inadequate proportional heaters appear to be energized
- Pressurizer Pressure is 2230 psig.

Which ONE of the following could result from inadequate Pressurizer Heater output during a dilution to criticality?
(Assume all other controls and components working properly in their normal configuration.)

A. Boron concentration will be higher in the Pressurizer than in the RCS.

B. Boron concentration will be lower in the Pressurizer than in the RCS.

C. Pressurizer and RCS boron concentration will be approximately equal.

D. The Pressurizer Spray Nozzle will be susceptible to thermal shock.

Surry

References:
1-GOP-1.1, Unit Startup, RCS Heatup From Ambient to 195 Degrees F, Rev. 25

Distractor Analysis:
A. Correct because RCS boron will be lower due to the dilution. The Pzr will still be at a higher boron concentration until spray flow has created enough out-surge to adequately equalize the boron with the RCS. (Lack of heaters creates lack of sprays.)
B. Incorrect because boron concentration will be higher in the Pzr.
C. Incorrect because the lack of heater output will not allow for adequate mixing.
D. Incorrect because the bypass spray valves are normally open, which is sufficient to prevent thermal shock. [Have utility verify that this is in fact the normal configuration.) This is correct

010 Pressurizer Pressure Control
A1.01: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Pzr PCS controls including: PZR and RCS boron concentration.
Answer: **A**

11. 011EA1.13 1

Given the following conditions:
- LOCA has occurred
- RCS subcooling is 0 ^{o}F
- RWST Level = 15% and slowly decreasing
- Containment Pressure = 9 psig and decreasing
- Safety Injection Actuation has been reset

Which ONE of the following is the correct action to be taken?

A. Close Charging Pump Miniflow Recirc Valves. With RWST level at 15%, push both RMT pushbuttens for each train if automatic transfer does not occur.

B. Close Charging Pump Miniflow Recirc Valves. When RWST level reaches 13%, push both RMT pushbuttons for each train if automatic transfer does not occur.

C. With RWST level at 15%, push both RMT pushbuttons for each train if automatic transfer does not occur. Secure Containment Spray Pumps immediately following verification of Phase 1 and 2 RMT.

D. With RWST level at 13%, push both RMT pushbuttons for each train if automatic transfer does not occur. Secure Containment Spray Pumps immediately following verification of Phase 1 and 2 RMT.

References:
ND-95.3-LP-7, E-I Loss of Reactor or Secondary Coolant, Rev. **14**
1-E-1, **Loss** of Reactor or Secondary Coolant, Rev. 21
1-ES-1.3, Transfer *to* Cold Leg Recirculation, Rev. 12

Distractor Analysis:
A. Incorrect because RMT transfer should not occur at 15%. It will occur at 23.5% and procedurally should be verified at 13%.
B. Correct because verifications should be made at 13% and manually initiated if needed (directed by ES-1.3).
C. Incorrect per ES-1.3 spray pumps should not be secured until they shew signs of cavitation. Also E-1 does not call for spray to be secured until containment pressure is less than 12 psia. Also, RMT transfer should not occur at 15%.
D. Incorrect per ES-1.3 spray pumps should not be secured until they show signs of cavitation. Also E-1 does not call for spray to be secured until containment pressure is less than 12 psia.

Surry ILT Bank Question # 872

011 Large Break LOCA
EA1.13: Ability to operate and monitor the following as they apply to a large break LQCA: Safety Injection Components.
Answer: B

12. 011K6.06 1

Due to a controller failure, the Unit 1 Operator places the Charging Flow Controller to MANUAL to control charging flow. A high Pressurizer Level causes the Operator to try to reduce charging flow to 20 gpm.

Which ONE of the following correctly describes the behavior of FCV-1122 when the Operator attempts to reduce charging flow to 20 gpm?

A. The Flow Limit Summator no longer limits flow and FCV-1122 can be manually closed to allow 20 gpm flow.

B. The Flow Limit Summator no longer limits flow, however, FCV-1122 can only be manually closed to allow 25 gprn flow.

C. The Flow Limit Summator will prevent FCV-1I22 from being closed past 25 gpm flow.

D. The Flow Limit Summator will prevent FCV-1122 from being closed past 30 gpm flow.

Surry

References:
ND-93.3-LP-7, Pressurizer bevel Control System. Rev. 6

Distractor Analysis:
A. Correct because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. Therefore FCV-1122 can be closed manually to any value.
B. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. Distractor is incorrect because FCV-1122 may be manually closed to any value, even below 25 gpm flow. Distractor is plausibe because candidate may not know that FCV-1122 may be throttled to any value with controller in MANUAL.
C. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. The distractor states that the Flow Limit Summator wilt limit flow, which *is* contrary to the fact that it will not limit Row. Distractor is plausible because candidate may not know that the Flow Limit Summator does not function with controller in MANUAL.
D. Incorrect because when the Charging Flow Controller is in MANUAL, the Flow Limit Summator no longer limits the maximum and minimum values of charging. The distractor states that the Flow Limit Summator will limit flow, which is contrary to the fact that it will not limit flow. Distractor is plausible because candidate may not know that the Flow Limit Summator does not function with controller in MANUAL.

611 Pressurizer Level Control
K6.06: Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Correlation of demand signal indication on charging pump flow valve controller to the valve position.
Answer: A

13. **01284.04** 1

"A" Loop Narrow Range Tcold fails low while the reactor is at 100%.

Which ONE of the following will occur?

A. Rod Insertion Limit Low and Extra Low alarms will be received.

B. Ch 1 OTDT setpoint will decrease.

C. "A" Loop OP/OT Delta T Protection Bistables will trip.

D. The Tavg / Tref Deviation alarm will be received.

Surry

References:
MD-93.3-LP-2, Delta T / Pavg Instrumentation System, Rev. 9
ND-93.3-LP-3, Rod Control System, Rev. **14**

Distractor Analysis:
A. Incorrect because failed Tcold is filtered out by Median Signal Selector.
B. Incorrect because OTDT setpoint will actually increase.
C. Correct because Tcold is fed directly to the RPS even when failed low.
D. Incorrect because failed Tcold is filtered out by Median Signal Selector.

012 Reactor Protection System
A4.04: Ability to manually operate and / or monitor in the control room: Bistables, trips, resets, and test switches.
Answer: *C*

14. 012K1.05 1

Which ONE of the following lists the method by which AMSAC causes a reactor trip?

A. Tripping the reactor trip and bypass breaker shunt coils.

B. Tripping the reactor trip and bypass breakers UV coils.

C. Tripping the rod drive MG set output breakers.

D. Tripping the rod drive MG set supply breakers.

Surry

References:
ND-93.3-LP-17, Anticipatory Mitigating System Actuating Circuitry, Rev. 10

Distractor Analysis:

A. Incorrect because this does not occur.
~~A.~~B. Incorrect because this does not occur.
~~A.~~C. Incorrect because this does not occur.
~~A.~~D. Correct becasue this is as stated in ND-93.3-LP-IT, Rev. 10.

012 Reactor Protection System
K1.05: Knowledge of the physical connections and / or cause-effect relationships between the RPS and the following systems: **ESFAS**

Answer: D

15. 013A3.02 1

Which ONE of the following correctly states automatic actions that would occur given a Unit 1 Low Pressurizer Pressure Safety Injection Signal being present **for** 5 minutes?

A. Hydrogen Analyzer Heat Tracing energizes AND Containment Vacuum Pumps trip.

B. Pressurizer Liquid Sample (SS-TV-100A) receives a close signal AND Motor Driven AFW Pumps start after a 45 second time delay.

C. Accumulator Nitrogen Relief Lines (SI-TV-101A,B) receive **e** close signal AND Primary Brain Transfer Tank Vents (VG-TV-109A/B) receive a close signal.

D. Main Steam Trip Valves (MS-TV-101A/B/C) receive a close signal AND Seal Water Return Valve (MOV-381) receive a close signal.

Surry

References:
ND-91-LP-2, Safety Injection System Description, Rev. 16
NB-91-LP-2, Safety Injection System Operations, Rev. 15
P&ID 11448-FM-068A, *Flow/Valve* Operating Numbers Diagram Feedwater System Surry Power Station Unit 1, Rev. 57

Distractor Analysis:
A. Incorrect because SI signal must be present for 8 minutes for heat trace to energize.
5. Incorrect because MDAFW Pump starts after 58 sec delay.
C. Correct because both get a close signal on any SI Signal.
D. Incorrect because MSTVs only get a close signal on a High Steam Flow SI Signal.

023 Engineered Safety Features Actuation
A3.02: Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment.

Answer: C

16. 013K3.01 1

Which ONE of the following could occur if ES-1.4, Transfer to Hot Leg Recirculation, is performed 20 hours after the start of a barge Cold Leg Break LOCA?

A. Debris from the In-Core sump could block coolant flow by blocking the lower core plate

B. Reflux cooling could be lost due to boron precipitation in the hot leg nozzles.

C. Fouling of core heat transfer surfaces due to the dilution of boric acid.

B. Reduction in size of the incore coolant flow channels due tu boron precipitation.

Surry

References:
ND-95.3-LP-11, ES-1.4, Transfer To Hot Leg Recirculation, Rev. 8
E§-1.4, Transfer To Hot beg Recirculation, Rev. 4

Distractor Analysis:
A. Incorrect because debris in the sump will not block water discharged from the SI pumps.
B. Incorrect because boron precipitation is a concern in the core, not the hot legs.
C. Incorrect because fouling of core heat transfer surfaces is a result of boron precipitatiun, not dilution.
D. Correct because boron precipitation is a concern when boil-off continues and when core temperature decreases. The standard time for transfer to hot leg recirc is 8 hours, not 20 hours, as stated in the stem.

013 Engineered Safety Features Actuation
K3.01: Knowledge of the effect that a loss or malfunction of the ESFAS wilt have on the following: Fuel

Surry Requal Bank Question #299

Answer: D

17. 044A2.05 1

The following Unit 1 conditions exist:
- A Small Break LOCA has occured
- Automatic Safety Injection has occurred
- I-E-0, Reactor Trip or Safety Injection, has been implemented
- The CRO observes the Rod Position Indication as displaying Control Rods on the bottom of the reactor core, with the exception of three Control Rods.

Which ONE of the following actions is procedurally required as a result of this finding by the CRO?

A. Continue with I-E-8, Reactor Trip or Safety Injection.

B. Emergency borate while proceeding through 1-E-0, Reactor Trip or Safety Injection.

C. Manually insert control rods while proceeding through I-E-0, Reactor Trip or Safety Injection.

D. Go directly to 1-FR-S.1, Response to Nuclear Power Generation / ATWS, Step I

Surry

References:
1-FR-S.1, Response to Nuclear Power Generation / ATWS, Rev. 18
I-E-0, Reactor Trip or Safety Injection, Rev. **46**

Distractor Analysis:
A. Correct because E-0 should be entered upon Reactor Trip per the rules of EOP usage.
B. Incorrect because if emergency boration is needed, it will be directed by FR-S.1.
C. Incorrect because if manual **rod** insertion is needed, it will **be** directed by FR-S.1.
D. Incorrect because FR-S.1 should only be entered as directed by E-0 (or if E-0 has been completed then an Orange or Red path).

014 Rod Position Indication
A2.05: Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor Trip.

Surry ILT Bank Question #1037
Answer: **A**

18. 015K4.07 1

The following Unit 1 conditions exist:
- Reactor Power is 5%
- Turbine First Stage Impulse Pressure PT-446 is selected
- Power Range Nuclear Instrument N-41 fails high
- PT-446 fails high

Which ONE *of* the following correctly describes the impacts of the failures?

A. Control Rods do not move. The Reactor Protection System At-Power Trips are enabled due to the N-41 failure.

B. Control Rods step out at 72 steps per minute. The Reactor Protection System At-Power Trips are enabled due to the N-41 failure.

C. Control Rods do not move. The Reactor Protection System At-Power Trips are enabled due to the PT-446 failure.

D. Control Rods step out at 72 steps per minute. The Reactor Protection System At-Power Trips are enabled due to the PT-446 failure.

Surry

References:
ND-93.3-LP-16, Permissive/Bypass/Trip Status Lights, Rev. 8
Surry Simulator Malfunction Cause and Effects, Rev. 6, Malfunction MMS-14

Distractor Analysis:

A. Incorrect because Tref will go to 574 ^{o}F, which will cause rods to step out at max rate of 72 steps/min. *Also* incorrect because 2/4 PR NIs must be above 10% to enable At Power Trips.
B. Incorrect because 2/4 PR NIs must be above 10% to enable At Power Trips.

C. ~~Incorrect because Tref will go to 574 ^{o}F, which will cause rods to step out at max rate of 72 steps/min.~~ Correct because rods are in manual at 5% power and will not move. **At** power trips are enabled when 1 of 2 Pimp channels goes above 10%.

D. ~~Correct because Tref will go to 574 ^{o}F, which will cause rods to step out at max rate of 72 steps/min and only 1/2 Turbine First Stage PTs need to be above 10% to enable At Power Trips.~~ Incorrect because rods are in manual at 5% pewer and will not move

015 Nuclear Instrumentation
K4.07: Knowledge of NIS design feature(s) end / or interlock(s) provide for the following: Permissives.
Answer: ~~D~~ C

19. 016A4.01 1

The following condition exists:
- Unit 1 at 100% reactor power
- All systems and equipment functions as designed
- All protection channel III's are selected
- First stage impulse pressure channel IV fails low

Which ONE of the following would occur initially without operator action?

A. AMSAC would be operationally disabled after 60 seconds.

E. Steam Dumps would all open.

C. FRVs would control SG level at no load level.

D. MOV-CP-100, Condensate Polishing Building Bypass Valve, would open.

Surry

References:
ND-93.3-LP-17, Anticipatory Mitigating System Actuating Circuitry (AMSAC), Rev. 10
ND-93.3-LP-9, Steam Dump Control System, Rev. 10
ND-93.3-LP-8, SG Water bevel Control System, Rev. 6

Distractor Analysis:
A. Incorrect because this would occur after 360 seconds.
B. Incorrect because Channel III is selected.
C. Incorrect because Channel III is selected.
D. Correct because, as stated in ND-93.3-LP-9, CP-100 will open in anticipation of the upcoming increase in feedwater flow that will occur during load rejection.

016 Non-Nuclear Instrumentation
A4.01: Ability to manually operate and / or monitor in the control room: NNI channel select controls.

Surry Requal Bank Question #279
Answer: D

20. 022AK1.01 1

The following Unit 1 conditions exist:
- Reactor trip has occurred due to a loss of all AC power
- Power has been restored
- The following Reactor Coolant Pump parameters are present for all RCPs:
  - NO. 1 Seal Water Outlet Temperatures are ~~225~~ 210 °F
  - Lower Seal Water Bearing Temperatures are ~~220~~ 190 °F
- The Shift Supervisor directs the operators to restore cooling to the RCP seals per 1-AP-9.02, Loss of RCP Seal Cooling.

Which ONE of the following correctly states the requirements for resturing cooling to the RCP seals and why?

A. Do not establish seal injection flow or component cooling flow to the thermal barrier heat exchanger because the No. 1 Seal Water Outlet Temperatures are too high. Seal cooling should be restored by cooling the RCS using natural circutation.

B. Do not establish seal injection flow or component cooling flow to the thermal barrier heat exchanger because the Lower Seal Water Bearing Temperatures are too high.
Seal cooling should be restored by cooling the RCS using natural circulation.

C. Slowly establish seal injection flow to minimize RCP thermal stresses, followed by slowly introducing component cooling flow to the thermal barrier heat exchanger to limit introduction of steam into the CC system.

D. Slowly establish component cooling flow to the thermal barrier heat exchanger to limit introduction of steam into the CC system, followed by slowly introducing seal injection flow to minimize the RCP thermal stresses.

Surry

References:
1-AP-9.02, Loss of RCP Seal Cooling, Rev. 8.
ND-88.1-LP-6, Reactor Coolant Pumps, Rev. 16.

Distractor Analysis:
A. Incorrect because AP-9.02 (Caution page 7) states if No. **1** Seal Water Outlet Temp is > 235 °F then Seal Inj and CCW to Thermal Barrier H.X. should not be restored. Instead N.C. should be used to cool the seals.
B. Incorrect because AP-9.02 (Caution page 7) states if Lower Seal Water Bearing Temperature is > 225 °F then Seal Inj and CCW to Thermal Barrier H.X. should not be restored. Instead N.C. should be used to cool the seals.
C. Incorrect because CC flow should be established prior to seal injection flow.
D. Correct as stated in I-AP-9.02 NOTE prior to step 7 and CAUTIONS prior to steps 9 and 15.

022 Loss of Rx Coolant Makeup
AK1.01: Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: Consequences of thermal shock to RCP seals.
Answer: D

Values for seal water outlet temp and lower seal water bearing temp are too close to the limits for these parameters. We do not require Operators to memorize values of this nature when they are readily available in the procedures.

If you change question as indicated, the candidates should recognize that these values are below the limits to allow **seal cooling** restoration and that is what you are testing.

21. 022G2.4.22 1

Unit 1 has tripped and Safety Injection has actuated due to a barge Break Loss of Coolant Accident (LOCA).

Many complications have occurred.

The crew has exited E-0, Reactor Trip or Safety Injection. The Shift Technical Advisor has started to monitor Critical Safety Function Status Trees and reports:
- Subcriticality - Orange Path
- Heat Sink -Yellow Path
- Core Cooling - Orange Path
- Containment - Red Path

Which ONE of the following states the correct procedure transition?

A. FR-S.1, Response to Nuclear Power Generation/ATWS, based on Subcriticality Orange Path

B. FR-H.1, Response to Secondary Heat Sink, based on Heat Sink Yellow Path.

C. FR-C.1, Response to Inadequate Core Cooling, based on Core Cooling Orange Path.

D. FR-Z.1, Response to High Containment Pressure. based on Containment Red Path.

Surry

References:
ND-95.3-LP-26, Critical Safety Function Status Trees, Rev. 5

Distractor Analysis:
A. Incorrect based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Subcriticality Orange Path does not take priority over any Red Path.
B. Incorrect based on the rules of use for safety function status trees (NB-95.3-LP-26 Page 15). The Heat Sink Yellow Path does not take priority over Containment Red Path.
C. Incorrect based on the rules of use for safety function status trees (NB-95.3-LP-26 Page 15). Core Cooling Orange Path does not take priority over Containment Red Path.
D. Correct based on the rules of use for safety function status trees (ND-95.3-LP-26 Page 15). The Containment Red Path takes priority over the other paths. Only knowledge of safety function priority rules are needed to answer this question.

022 Containment Cooling
G2.4.22: Knowledge of the bases for prioritizing safety functions during abnormal and emergency operations.

Turkey Point Bank Question TP03301
Answer: D

22. 022K3.02 1

Unit 2 is operating at 100% power with Chilled CC in service to containment. 2-CD-REF-1% Unit 2 Turbine Building Chiller Unit, trips due to a fault.

Which ONE of the following describes the effect on Unit 2 containment parameters?

A. Indicated partial pressure will increase. Containment temperature will decrease.

B. Indicated partial pressure will increase. Containment temperature will increase.

C. Indicated partial pressure will decrease. containment temperature will decrease.

D. Indicated partial pressure will decrease. Containment temperature will increase.

Surry (Utility should add noun names to equipment in the stem.)

References:
ND-88.5-LP-1, Component Cooling, Rev. 19

Distractor Analysis:
A. Incorrect because partial pressure will decrease due to loss of chilled CC.
B. Incorrect because partial pressure will decrease due to loss of chilled CC.
C. Incorrect because containment temperature will increase due to a toss of chilled CC.
D. Correct because partial pressure will decrease and containment temperature will increase due to a loss of chilled CC.

Bank Question # 544

022 Containment Cooling
K3.02: Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment Instrument Readings.
Answer: D

23. 026A2.07 1

The following Unit 1 conditions exist:
- *A* Large Break LOCA has occurred inside containment
- Safety Injection has actuated
- Containment Pressure peaked at **28** psia
- Current Containment Pressure is 15.8 psia
- ~~"1A",~~ "2A" and "2B" Outside Recirculation Spray Pumps are operating
- "1A" Inside Recirculation Spray Pump is operating
- "1B" Inside Recirculation Spray Pump tripped on Overload (OL)
- 1A-E7, RS PP 1A VIB, annunciates and the alarm cannot be cleared

Which ONE of the following states the correct operator action for these conditions?

A. Secure Inside Recirculation Spray Pump "1A" using the handswitch in *the* control *room.*

B. Place the Inside Recirc Spray Pump I A in PTL, then secure Inside Recirculation Spray Pump "1A" locally at the breaker (14H4).

C. Reset CLS, then place the handswitch for Inside Recirculation Spray Pump "1A" in PTL.

D. Allow Inside Recirculation Spray Pump "1A" to operate, *but* monitor vibrations closely.

Surry

References:
ND-91-LP-5, Containment Spray System, Rev. 13
ND-91-LP-6, Recirculation Spray System, Rev. 9
1-RM-A7, RS/SW HX *A* ALERTFAILURE, Rev. 5

Distractor Analysis:
A. Incorrect because with CLS present, the handswitch in the control room cannot be used to secure the pump. Containment pressure must be less than 12 psia to reset CLS. Pressure currently **is** 15.8 psia.
B. Correct because local operation of the breaker will stop the pump. In addition, the ARP will have the operator place the handswitch in PTL, but the lesson plan (ND-91-LP-6 Page *6)* states that the pump cannot be secured from the control room with CLS present. Furthermore. the ARP gives guidance to secure the distressed pump as long as two other RS Pumps are operating. The stem states that two other pumps are operating ("2A" and "2B").
C. Incorrect because the CLS cannot be reset until containment pressure is less than 12 psia.
D. Incorrect because the ARP gives guidance to secure the distressed pump as long as two other RS Pumps are operating. The stem states that two other pumps are operating ("2A" and "2B").

026 Containment Spray
A2.07: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequnces of these malfunctions or operations: Loss of containment spray suction when in recirculation mode, possibly caused by clogged sump screen. pump inlet high temperature (exceeded cavitation, voiding). or sump level below cutoff (interlock) limit.

Note:
The ARP states that high vibration alarms may be caused by cavitation of the pump. Cavitation could be caused by high water temp, low water level, etc.

Answer: B

24. 026AK3.02 1

A High Steam Flow Safety Injection Signal is received.

Which ONE of the following correctly describes the response **of** the Component Cooling Water System components?

A. TV-CC-109A and B (CC Isolation Valves from RHR) close and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trip Valve) remain as-is.

B. TV-CC-109A and B (CC Isolation Valves from RHR) remain as-is and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler 66 Outlet Flow Outside Trip Valve) remain as-is.

C. TV-CC-109A and B (CC Isolation Valves from RHR) close and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trip Valve) close.

D. TV-CC-109A and B (CC Isolation Valves from RWR) remain as-is and TV-CC-110A, B, and C (Reactor Cont Air Recirc Cooler CC Outlet Flow Outside Trip Valve) dose.

Surry

References:
88-05-04, Component Cooling Water System, Rev. 49

Bistractor Analysis:
A. Correct because lesson plan states CC-109 closes on Phase I and 110 closes on Phase III isolation.
B. Incorrect because lesson plan states CC-109 closes on Phase I and 110 only closes on Phase III isolation.
C. Incorrect because lesson plan states CC-109 closes on Phase I and 110 only closes on Phase III isolation.
D. Incorrect because lesson plan states CC-109 closes on Phase I and 110 closes on Phase III isolate.

026 Loss of Component Cooling
AK3.02: Knowledge of the reasons for the following responses as they apply to Loss of Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS.

The loss of CCW occurs in part of the system **due** to the ESFAS isolation of TC-CC-109A/B
Answer: **A**

25. 026G2.4.46 1

The following Unit 1 conditions exist:
- A Large Break LOCA ~~has occurred~~ 45 minutes ago
- Safety Injection has actuated
- Containment Pressure peaked at 27 psia
- RCS subcooling is 0 ^{o}F
- Steam Generator bevels are 22% and slowly rising
- RWST emptied while performing ES-1.3, Transfer to Cold Leg Recirculation
- ES-1.3, Transfer to Cold Leg Recirculation, has been completed and the crew has transitioned back to E-l, Loss of Reactor or Secondary Coolant
- All equipment operated normally

Which ONE of the following alarms is consistent given the above plant conditions?

A. 1E-AI. HI-HI CTMT PRESS CLS CH-1

B. 1B-B1, CS PP 1A LOCKOUT OF?OL TRIP

C. 1A-D7, RS PP ~~1A~~ 1A LOCKOUT OR OL TRIP (OK as is)

D. 1B-F6, CTMT INST AIR HDR LO PRESS

Surry

References:
I-€-I, Loss of Reactor or Secondary Coolant, Rev. 21
I-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 12
IE-A1, HI-HI CTMT PRESS CLS CH-1, Rev. 0
48-81, CS PP 1A LOCKOUT OF?Ob TRIP, Rev. 0
1/4-57, RS PF 1A LOCKOUT OR OL TRIP, Rev. 0
1B-F6, CTMT INST AIR HDR LO PRESS, Rev. 1
ND-91-LP-5, Containment Spray System, Rev. I 3
ND-91-LP-6, Recirculation Spray System, Rev. 9

Distractor Analysis:
A. Incorrect because containment pressure is now less than the setpoint, which is known by CLS having been reset. As a part of going to Cold Leg Recirc, GLS and SI must be reset.
B. Currect because 4-ES-1.3 has been completed and the RWST has been emptied; therefore, the CS Pumps would be placed in PTL due to the lack of a suction source (cavitation). Placing the CS Pumps in PTL yields 1B-B1 for CS Pump 1A.
C. Incorrect because ~~Outside~~ Inside Recirc Spray Pump ~~1A~~ 1A would be placed in AUTO when stopped.
D. Incorrect because CLS and SI must have been reset prior to completion of 1-ES-I.3 and instrument air would have been restored to containment.

026 Containment Spray
G2.4.46: Ability to verify that alarms are consistent with plant conditions.
Answer: B

Adding 45 minutes ago *to* the stem of the question ensures that the trainees know the containment has had time to depressurize below GLS reset point to allow Inst Air to be realigned.

26. 027A4.03 1

Which ONE of the following describes the operation of the Iodine Filtration Fans (1-VS-F-3A/3B)?

A. Automatically start on a Hi-Hi CLS.

B. Automatically start on a containment gas high alarm.

C. Automatically stop on a Hi-Hi CLS signal.

D. Must be manually started under all conditions.

Surry

References:
ND-88.4-LP-6, Containment Ventilation, Rev. 5

Distractor Analysis:
A Incorrect because fans are only manually operated.
B. Incorrect because fans are only manually operated.
C. Incorrect because fans are only manually operated.
D. Correct because fans are only manually operated.

027 Containment Iodine Removal
A4.03: Ability *to* manually operate and / or monitor in the control room: CIRS fans

Question Status:
Surry Bank ILT Question #741
Answer: D

27. 027AK3.01 1

The following Unit 1 conditions exist:
- The Reactor is at 100% Power.
- A malfunction in the Pressurizer Heater Control Circuit has resulted in Proportional Heaters being de-energized.
- A small amount of leakage in the Pressurizer Auxiliary Spray Valve is occurring.
- Pressurizer Pressure is 2215 psig and slowly lowering.

1-AP-31.00, Increasing or Decreasing RCS Pressure, has been entered.

Which ONE of the following states the correct position of the normal sprays and backup heaters?

A. Normal sprays are OFF (valves closed) and backup heaters are ON.

B. Normal sprays are ON (valves open) and backup heaters are OFF.

C. Normal sprays are OFF (valves closed) and backup heaters are OFF.

D. Normal sprays are ON (valves open) and backup heaters are ON.

Surry

References:
ND-93.3-LP-5, Pressurizer Pressure Control, Rev. 9
IC-88, PRZR LO PRESS, Rev. 1
1-AP-31.00, Increasing or Decreasing RCS Pressure, Rev. 6

Distractor Analysts:
A. ~~Incorrect~~ because backup heaters are normally on ~~do not energize until 2210 psig.~~
B. Incorrect because spray valves do not start to open until 2255 psig.
C. ~~Correct~~ Incorrect because backup heaters ~~do not energize until 2210 psig~~ are normally on and spray valves do not open until 2255 psig.
D. Incorrect becaus~~e backup heaters do not energize until 2210 psig~~ because normal spray valves are OFF (valves closed).

027 Pressurizer Pressure Control System Malfunction
AK3.01: Knowledge of the reasons for the following responses **as** they apply to pressurizer pressure control malfunctions: Isolation of PZR spray following loss of PZR heaters.
Answer: ~~C~~ A

The pressurizer heater control system does not operate as originally designed. The Backup Heater Control Switches are maintained in the "ON" position instead of "AUTO".

28. 028G2.2.12 1

The following Unit 1 conditions exist:
- The plant is at 50% power
- 1-PT-37.2, Electric Hydrogen Recombiner, is about to be performed to determine the reference power that would be used in the event that the Recombiners are used following a LOCA.

Which ONE of the following correctly states 1-PT-37.2 limitations that are applicable during the performance of this test?

A. At no time should the heater temperature be allowed to exceed 4400 °F as monitored by the highest thermocouple reading AND containment hydrogen concentration must be verified to be less than 0.75%.

B. At no time should the heater temperature be allowed to exceed 1400 °F as monitored by the highest thermocouple reading AND containment hydrogen Concentration must be verified to be less than 1.00%.

C. At no time should the heater temperature be allowed to exceed 1300 °F as monitored by the highest thermocouple reading AND containment hydrogen concentration must be verified to be less than 0.75%.

D. At no time should the heater temperature be allowed to exceed 1300 °F as monitored by the highest thermocouple reading AND containment hydrogen concentration must be verified to be less than 1.00%.

Surry

References:
1-PT-37.2, Electric Hydrogen Recombiner, Rev. 9

Distractor Analysis:
A. Correct because these are both requirements listed on Section 4.0 of I-PT-37.2. The unit is at power, therefore 4.3 states that containment hydrogen concentration must be verified less than 0.75% (being at power and making the operator determine if 4.3 applies is part of what makes the question C/A). Section 4.2 states that heater temperature must remain less than 1400 °F at all times.
B. Incorrect because verifying containment hydrogen less than 1% is not the currect requirement. Plausible because applicant may not know that the requirement is 0.75%. vice 1.0%.
C. Incorrect because verifying the highest temperature less than 1300 °F is not the correct requirement. True - if the operator ensures temperature is less than 1300 °F, then he has also ensured that it is less than 1400 °F, but this question tests the knowledge of the requirement, not simply a method for meeting the requirement. Plausible because applicant may not know the temperature requirement.
D. Incorrect because of reasons in C and 5 distractor analysis.

028 Hydrogen Wecombiner and Purge Control
G2.2.12: Knowledge of surveillance procedures.
Answer: A

The information being tested in this question for containment hydrogen concentration is not required knowledge for an Operator. Recommend changing question choices to read as follows:

a. Turn Power Adjust potentiometer to obtain 48 KW on wattmeter and approximately 1225 ^{o}F heater output. At no time should the heater temperature be allowed to exceed 1400 ^{o}F on the highest thermocouple reading.

b. Turn Power Adjust potentiometer to obtain **48** KW on wattmeter and approximately 1000 ^{o}F heater output. At no time should the heater temperature be allowed to exceed 1200 ^{o}F on the highest thermocouple reading.

c. Turn Power Adjust potentiometer to obtain 36 KW on wattmeter and approximately 1225 ^{o}F heater output. **At** no time should the heater temperature be allowed to exceed 1400 ^{o}F on the highest thermocouple reading,

d. Turn Power Adjust potentiometer to obtain 36 KW on wattmeter and approximately 1000 ^{o}F heater output. At no time should the heater temperature be allowed to exceed 1200 ^{o}F on the highest thermocouple reading.

Correct Answer (A)

29. 029EK3.09 1

Which ONE of the following describes the reason why charging pump suctions are manually aligned to the RWST during an ATWS vice manually initiating a Safety Injection?

A. Prompt operator action will ensure the most direct method of borating into the RCS and manual alignment of charging pump suction to the RWST prevents compounding the probiem by charging the RCS solid via Safety injection.

B. Prompt operator action will ensure the most direct method of borating into the RCS and initiation of SI would reduce the possible paths for emergency boration end add to an RCS overpressure condition if one exists.

C. Manual initiation of Safety Injection would delay the addition of borated water to the RCS and complicate the recovery actions. Alignment of charging pump suction to the RWST is the most direct method of borating the RCS.

D. Manual initiation of SI would result in the undesirable trip of Main Feedwater Pumps and alignment of Charging Pump suction to the RWST is the most direct method of borating the RCS.

Surry

References:
ND-95.3-LP-36-DRR, FR-S.1 Response to Nuclear Power Generation / ATWS, Rev. 10
FR-S.1, Response to Nuclear Power Generation / ATWS, Rev. 15

Distractor Analysis:
A. Incorrect because the concern with initiating SI is not creating a solid plant condition, but with reducing the probability of maintaining a secondary heat sink because MFW pumps will trip upon SI initiation.
B. Incorrect because the concern with initiating SI is not creating a high RCS pressure condition, but with reducing the probability of maintaining a secondary heat sink because MFW pumps will trip upon SI initiation.
C. Incorrect because manual initiation would not delay addition of borated water. The concern is with reducing the probability of maintaining a secondary heat sink because MFW pumps will trip upon SI initiation.
D. Correct because per ND-95.3-LP-36-DRR, FR-S.1 Response to Nuclear Power Generation / ATWS, both of these statements accurately reflect the basis for Step 4.

029 ATWS
EK3.09: Knowledge of the reasons for the following responses as they apply to the ATWS: Opening centrifugal charging pump suction valves from RWST.

Modified ILT Bank Question # 3390
Answer: D

30. 032AA1.01 1

The following conditions exists:
- Present time is 1428 hours
- Reactor tripped at 1405 hours
- All Rod Bottom Lights are lit
- N-35 reading is 2 x 10^{-10} amps
- N-36 reading is **4** x 10^{-11} amps
- Source Range ***is*** nut energized
- Power level prior to trip was 90%

Which ONE of the following describes the correct actions given the above parameters?

A. When both IR channels read < 5 x 10^{-10} amps, verify source range channels energized.

B. Place the source range trip bypass switches in the NORMAL position.

C. Energize the source range channels by depressing the source range manual reset pushbuttons.

B. Transfer NR-45 to one source range and one intermediate range channel

Surry

References:
ND-93.2-LP-3, Intermediate Range NIs, **Rev.** 10.

Distractor Analysis:
A. Incorrect because SR energizes at 2/2 IR < 5 x 10^{-11} amps.
B. Incorrect because SR should already be energized in the NORMAL position and **this** action would not energize the SR.
C. Correct because IR are under-compensated and SR must be manually energized.
D. Incorrect because SR should both be energized.

032 Loss of Source Range NI
AA1.01: Ability to operate and / or monitor the following *as* they apply to loss of source range nuclear instrumentation: Manual restoration of power.
Answer: C

31. 033AA2.04 1

The following Unit 1 conditions exist:
- Critical approach has just been completed.
- Reactor is stable at the Point of Adding Heat.
- One Intermediate Range (IR) Nuclear Instrument (NI) is suspected of displaying inaccurate indications.

Which ONE of the following correctly describes the expected Power Range (PR) MI and the known operable IR NI indications for the above conditions to verify that the suspect IR NI is in fact falsely indicating?

A. IR = 2.5×10^{-8} Amps; PR between 0.2 and 1 %

B. IR = 2.0×10^{-6} Amps; PR between 0.2 and 1 %

C. IR = 1.0×10^{-8} Amps; PR < 0.2 %

D. IR = 1.0×10^{-5} Amps; PR < 0.2 %

Surry

References:
ND-93.2-LP-4, Power Range NIs, Rev. 16
1-GOP-1.4, Unit Startup, HSD to 2% Reactor Power, Rev. 29

Distractor Analysis:
A. Incorrect because 2.5×10^{-8} Amps is about where critical data is taken (too low).
B. Correct based on above two references: ND-93.2-LP-4 (H/T-4.3) *8* I-GOP-1.4 (Page 29 CAUTION).
C. Incorrect because 1.0×10^{-8} Amps is about where critical data is taken (too low).
D. Incorrect because 1.0×10^{-5} Amps is above the POAH and should correspond to about 2% power.

033 Loss of Intermediate Range NI
AA2.04: Ability tu determine **and** interpret the following as they apply to the loss of intermediate range nuclear instrumentation: Satisfactory overlap between source-range, intermediate-range, end power-range instrumentation.
Answer: B

32. 034A4.01 1

Unit 1 is in a refueling outage when the following events occur:
- Purge Isolation Valves (MOV-VS-100A, B, C, and D) Close
~~Unit Purge Supply Fans (4A and 4B) Trip~~
- containment Instrument Air Suction Valves (TV-IA-101 A/B) Close

Which ONE of the following radiation monitors could have caused these actions?

A. Process Vent Particulate and Gas Monitors (RM-RI-101 / 102)

B. RM-161 (Containment High Range Gamma Monitor)

C. RM-162 (Manipulator Crane Monitor)

B. RM-163 (Reactor Containment Area Monitor)

Surry

References:
ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 5

Distractor Analysis:
A. Incorrect because RM-RI-101 / 102 do not cause these actions.
5. Incorrect because RM-161 does not cause these actions.
C. Correct per ND-93.5-LP-1.
D. Incorrect because WM-163 does not cause these actions.

034 Fuel Handling Equipment
A4.01: Ability to manually operate and / or monitor in the control room: Radiation Levels.
Answer: C

Unit Purge Supply Fans never sperate – delete from stem of question

33. 035A3.01 1

The following Unit 1 conditions exist:
- Plant is stable at 75% Power
- "A" SG Steam bine FT-MS-475 (CH-IV) is selected for Steam Generator Level control
- " A SG Steam Line PT-MS-475 (CH-III) fails high

Which ONE of the following correctly describes the impact on the "A" Steam Generator Level control?

A. Feedwater Regulating Valve opens because indicated steam flow is greater than indicated feedwater flow.

B. Feedwater Regulating Valve does not move as a result of the failure.

C. Feedwater Regulating Valve closes because the pressure transmitter is overcompensating **for** density.

D. Feedwater Regulating Valve opens to reduce the level error created by the failure.

Surry

References:
ND-93.3-LP-8, SG Water bevel Control System, Rev. 6

Distractor Analysis:
A. Incorrect because PT-MS-475 does not cornpensate steam flow for FT-MS-475.
B. Correct because PT-MS-475 does not compensate steam flow for FT-MS-475.
C. Incorrect because PT-MS-475 does not compensate steam flow for FT-MS-475.
D. Incorrect because PT-MS-475 does not compensate steam flow for FT-MS-475.

035 Steam Generator
A3.01: Ability to monitor automatic operation of the S/G including: S/G water level control.
Answer: B

Trainees are not required to memorize mark numbers for most instrumentation. They normally deal with channel numbers. Recommend putting both mark numbers and channel numbers in stem.

34. 038EK3.09 **1**

Which ONE of the following is correct regarding safety injection termination during a steam generator tube rupture event?

Safety Injection termination ...

A. may occur with total AFW flow less than 350 gpm as long as 350 gpm is available.

B. may occur with Pressurizer level less than 22% as long as level is increasing.

C. may not ~~not occur~~ with **a** void in the reactor head due to presenting RCS pressure control problems.

D. may net &occur with a void in the reactor head due to presenting RCS level control problems.

Surry

References:
ND-95.3-LP-13, E-3 Steam Generator Tube Rupture: Rev. **11**
2-E-3, Steam Generator Tube Rupture, Rev. **19**

Distractor Analysis:
A. Correct because if no intact SG is available the ruptured SG will be used to cool the RCS. In this instance the **AFW** flow may be less than 350 gpm, but 350 gpm must still be available to that SG. If sufficient flow is available, then SI termination criteria is considered to be met (ND-95.3-LP-13).
B. Incorrect because pressurizer level must be greater than 22% to meet the SI termination criteria.
C. Incorrect because safety injection may be terminated when there is a void in the reactor head. This will present some challenges with RCS pressure and level control, but it is not a large enough concern to prevent SI termination if the specified criteria are met (ND-95.3-LP-13).
D. Incorrect because safety injection may be terminated when there is a void in the reactor head. This will present some challenges with RCS pressure and level control, but it is not a large enough concern to prevent SI termination if the specified criteria are met (ND-95.3-LP-13).

038 Steam Gen. Tube Rupture
EK3.09: Knowledge of the reasons for the following responses as they apply to the SGTR: Criteria for securing /throttling ECCS.
Answer: A

35. 039A1.09 1

With Unit 1 at 100% power, the Condenser Air Ejector and Main Steam bine Radiation Monitor alarms are recieved. The Condenser Air Ejector Radiation Monitor reads 700 cpm (ALERT and HIGH alarms are in) while local Main Steam NRC Radiation Monitors read "A" .03 mr/hr, and "B" .01 mr/hr, and "C" .01 mr/hr. The Team has implemented 1-AP-16.00, Excessive RCS Leakage, and the WCS leak rate is determined to be 60 gpm.

Which ONE of the following describes the actions required?

A. Verify automatic Condenser Air Ejector divert to Containment, intiate 1-AP-24.00 (Minor SG Tube beak), manually trip the reactor and go to 1-E-0 (Reactor Trip or Safety Injection).

B. Verify automatic SGBD TV trip isolation and Condenser Air Ejector divert to Containment, manually trip the reactor and initiate SI, Go to 1-E-0.

C. Verify automatic Condenser Air Ejector divert to Containment, initiate 1-AP-24.01 (Large Steam Generator Tube beak), verify letdown isolated, and commence a normal Unit shutdown IAW GOPs.

D. Verify automatic Condenser Air Ejector divert to Containment, initiate 1-E-0, Reactor Trip or Safety Injection and GO TO 1-AP-24.01 (barge Steam Generator Tube Leak)~~, and manually trip the reactor and go to 1-E-0~~.

The order of performance in 1-AP-16 is to trig the reactor first and then go to 1-AP-24.01.

Surry

References:
ND-89.3-LP-2, Main Condensate System, Rev. 16
NB-93.5-bP-3, Post-TMI Radiation Monitoring System, Rev. 6
1-AP-16.00, Excessive RCS Leakage, Rev. 41
1-AQ-24.00, Minor SG Tube Leak, Rev. 8
1-AP-24.01, Large Steam Generator Tube Leak, Rev. 11

Bistractor Analysis:
A. Incorrect because 60 gpm leakage is more than minor. AP-24.01 should be entered for a large steam generator tube leak.
B. Incorrect because SI should not be initiated.
C. Incorrect because the reactor must be manually tripped with leakage greater than 50 gpm.
D. Correct because air ejectors will divert to containment on an air ejector high radiation, AP-24.01 should be entered due to 60 gpm leak rate with air ejector high radiation, and E-0 should be entered following a manual reactor trip.

039 Main and Reheat Steam
A1.09: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main steam line radiation monitors.

Answer: D

36. 054G2.4.31 1

The following Unit 1 conditions exist:
- The Reactor was operating at 78% power when a loss of the "**A**" Feedwater Pump occurred
- The Team is taking the required immediate actions in acordance with 1-AQ-21.00, "Loss of Main Feedwater Flow"
- The Reactor Operator is driving rods in manual to lower Tavg
- Tavg is within 3 °F of Tref
- Annunciator 1G-G8, ROB BANK D LO LIMIT, has annunciated

Which ONE of the following is the correct response **to** the given plant conditions?

A. Shutdown margin is not sufficient for the given plant conditions and operators should emergency borate to regain the required shutdown margin.

B. The operator has driven rods in too far for the existing boron concenration and should borate from the Boric Acid Tanks.

C. Shutdown margin is not sufficient for the given plant conditions and operators should trip the Reactor and go to I-E-0, Reactor Trip or Safety Injection.

D. The turbine load has decreased too far and the operator should raise turbine load

Sur9

References:
ND-89.3-LP-3, Main Feedwater System, Rev. 12
ND-95.1-LP-4, Loss of Feedwater, Rev. 3
1-AP-21.00, Loss of Main Feedwater Flow, Rev. 5
1G-G8, ROD BANK D LO LIMIT, Rev. 0

Distractor Analysis:
A. Incorrect because (1) not enough information is given to make the determination that SDM is insufficient, and (2) even if SDM is not above that which is required, emergency boration would not be the preferred method for regaining the required SDM. This *is* clearly the wrong method for boration because xenon is building in and only small borations would be desired to withdraw rods to clear the alarm.
B. Correct because rods being within 10 steps of its insertion limit would cause the alarm. Boration from the Boric Acid Tanks would be the correct mitigation strategy and as such, is directed by the ARP. Operators would only borate the necessary amount to clear the alarm.
C. Incorrect because the initial power level was less than 85% and the plant is designed to handle this magnitude of transient. Furthermore, the plant does not need to be tripped with rods approaching or below insertion limits. Rod positions just have to be restored to within limits.
D. Incorrect because turbine load should net be raised. Immediate actions have the operators reduce turbine load to match steam flow and feed flow. Raising turbine load under these conditions would not be the correct action. It would also be nonconservtaive to add positive reactivity via the turbine during **a** transient condition such as described in the stem.

054 Loss of Main Feedwater
G2.4.31: Knowledge of annunciators and indications and use of response instructions.

Bank Question from 2003 Farley Exam (Farley K/A was 054G2.2.20).
Answer: B

37. 055EK1.01 1

The following plant conditions exist:
- A loss of all AC power has occurred.
- Operators have implemented ECA-0.0, Loss of All AC Power.
- Attempts to regain AC power have failed.
- Operators are performing ECA-0.0, Step 28, "Check DC Bus Loads"
- Annunciator J-F-6, Turb Gear Zero Speed, is lit

Which ONE of the following should be performed to lower the Black Battery discharge rate by the largest amount per ECA-0.0?

A. Secure Air Side Seal Oil Pump only.

B. Secure Air Side Seal Oil Pump and Emergency Turbine Lube Oil Pump

C. Secure Air Side Seal Oil Backup Pump only.

D. Secure Air Side Seal Oil Backup Pump and Emergency Turbine Lube Oil Pump.

Surry

References:
ND-90.3-LP-6, 125 Vdc Distribution, Rev. 10
ECA-0.0, Loss of All AC Bower, Rev. 21

Distractor Analysis:
A. Incorrect because the Air Side Seal Oil Pump is not a DC load, as is the Air Side Seal Oil Backup Pump. Plausible bemuse the candidate may not know major Black Battery DC Loads, or may not know what actions are permitted by ECA-0.0.
B. Incorrect because the Air Side Seal Oil Pump is not a DC load, as is the Air Side Seal Oil Backup Pump. Plausible because the candidate may not know major Black Battery DC Loads, or may not know what actions are permitted by ECA-0.0.
C. Incorrect ECA-0.0 will direct the securing of both Air Side Seal Oil Backup Pump (ASSOBUP) and Emergency Turbine Lube Oil Pump, not just ASSOBUP. Plausible because the applicant may not know that there is more than one pump to secure to conserve Black Batteries.
D. Correct because per ECA-0.0 step 28 and Basis for this step in ND-95.03-LP-17, the purpose is to secure both pumps, which are large Black Battery DC loads, to conserve the batteries (reducing battery discharge rate, thus prolonging battery life).

Surry ILT Bank Question #724

055 Station Blackout
EK1.01: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Effect of battery discharge rate on capacity.
Answer: D

Step 28 of ECA-0.0 has the operator verify annunciator J-F-6 lit before stopping the DC Emergencv Oil Pump.

38. 056A2.04 1

The following Unit 1 conditions exist:
- Two Main Feedwater Pumps are operating
- Reactor Power = 65%
- Condensate Pumps 1-CN-P-1A and B are operating
- Condensate Pump 1-CN-P-1C is Tagged Out of Service
- Condensate Pump 1-CN-P-1A trips and cannot **be** restarted
- Main Feedwater Pump Suction Pressure = 105 psig and slowly lowering
- Steam Generator Levels are slowly lowering
- 1H-F8, FW PP SUCT HBR LO PRESS, is in alarm

Which ONE of the following is the correct operator action?

A. Enter 1-AP-21.00, Loss of Main Feedwater Flow, and reduce turbine load to match steam flow and feedwater flow.

B. Manually trip the Reactor and enter E-0, Reactor Trip or Safety Injection.

C. Secure one of the operating Main Feedwater Pumps and monitor the operating Main Feedwater Pump Suction Pressure.

D. Enter 1-AQ-2100, Loss of Main Feedwater Flow, and start a second HP Drain Pump.

Surry

References:
ND-89.3-LP-2, Main Condensate System, Rev. 16
NB-89.3-LP-3, Main Feedwater System, Rev. 12
ND-95.1-LP-4, Loss of Feedwater, Rev. 3
4-AP-21.00, boss of Main Feedwater Flew, Rev. 5
1H-F8, FW PP SUCT HDR LO PRESS, Rev. 0
1H-G8, FW PP DISCH HDR LO PRESS, Rev. 0
1J-G4, CN PPS DISCH HDR LO PRESS, Rev. 0

Distractor Analysis:
A. Correct because MFW Pump Low Suction Pressure and Discharge Pressure Alarms are entry conditions into AP-21.00. Furthermore, with power at 65%, the direction is to reduce turbine load to match steam and feed flows. This will also help to recover MFW Pump suction/discharge pressure.
B. Incorrect because no trip criteria are met and AP-21.B0 directs power reduction.
C. Incorrect because tripping a MFW Pump will not alleviate the issue and there is no procedural guidance to trip a MFW Pump. Typically a MFW Pump will be secured at about 40% power.
D. Incorret because there is no guidance to start a second heater drain pump. The correct response is to lower turbine load.

056 Condensate
A2.04: Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: boss of condensate pumps.

Answer: **A**

39. 056AA1.26 1

The following conditions exist:
- A Loss of Off-Site Power has occurred
- #1 Emergency Diesel Generator has started but failed to auto load
- It has been determined that the auto-clusure circuit for 15H3, #1 EDG Output Breaker, is inoperable and that 15H3 can be manually closed
- When the operator places the sync switch for 15H3 to "ON" he observes 120 volts on the "incoming" meter, 0 volts on the "running" meter, and the synchroscope is stationary at "3-o'clock"

Which ONE of the fallowing actions is necessary prior to closing 15H3?

A. Raise EDG speed until the synchroscope is turning slowly in the fast direction, then close 15H3 at "11 o'clock".

B. Momentarily press the "field flash" pushbutton, then sync and close 15H3.

C. Raise EDG voltage until the running meter indicates 120 volts, then sync and close 15H3.

D. No additional action is necessary. Close 15H3.

Wave utility verify distractor analysis for B. I could not find anything that stated that field flash would need to be pushed under these conditions, but I want to ensure that the distractor analysis is correct.

References:
ND-90.3-LP-1, Emergency Diesel Generator, Rev. 14
ND-90.3-LP-7, Station Service and Emergency Distribution Protection and Control. Rev. 17

Distractor Analysis:
A. Incorrect because the bus is dead. Raising EDG speed will not synchronize the phases.
B. Incorrect because it will not be possible to synchronize (nor is it necessary because the bus is dead). Also, field flash PB does not need to be pushed.
C. Incorrect because raising the EDG voltage will not raise running voltage. incoming voltage *is* the EDG voltage (not running voltage).
D. Currect because the synchroscope has been turned on, there is nu over-current or differential and the aux trip relay does not need to be reset (ND-90.3-LP-7 pg. 18). Therefore, all criteria for manually closing the breaker are met.

056 Loss of Off-Site Power
AA1.26: Ability to operate and / or monitor the following as they apply to the Loss of Off-Site Power: Circuit Breakers
Answer: D

40. 057AK3.01 1

Which **ONE** of the following reasons correctly states why the reactor would be tripped for a sustained loss of Vital Bus II?

A. Power to the Reactor Protection System is lost.

B. Pressurizer pressure control is lost.

C. Control of Steam Generator Feed Regulating Valve is lost.

D. "B" Reactor Coolant Pump must be stopped.

Surry

References:
I-AP-10.02, Loss of Vital Bus II, Rev. 9
ND-90.3-LP-5, Vital and Semi-Vital Bus Distribution, Rev. 11

Distractor Analysis:
A. Incorrect because RPS is de-energize to trip. If due to other channel failures, etc., the loss of VB II will not preclude a trip if one is needed.
B. Incorrect because Pzr P Controller will transfer to AUTO-HOLD, but MANUAL control is still possible, thus precluding the need for rx trip.
C. Incorrect because FW-FCV-1488 Flow Controller will transfer to AUTO-HOLD, but MANUAL control is still possible, thus precluding the need for rx trip.
D. Correct because Component Cooling is lost to the "B" RCP Lube Oil Cooler. RCP Parameters will eventually exceed limits (1-AP-10.02 Att. 1) requiring that the RCP be secured following a manual rx trip.

057 Loss of Vital AC Inst. Bus
AK3.01: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical bus.

Surry ILT Bank Question #223
Answer: D

41. 058AA2.01 1

The following Unit 4 conditons exist:
- 1K-A8, UPS SYSTEM TROUBLE, annunciates
- 1K-A7, BATT SYSTEM 1A TROUBLE, annunciates
- An operator reports that Battery Charger DC Qutput for UPS 1A-1 reads 0 amps

Which ONE of the following correctly describes the power supply to the associated DC and Vital AC buses?

A. DC Bus 1A-4 will be supplied by only Battery 1A as indicated by DC Bus voltage slowly trending| down over time and Vital AC Buses I and 1A will be supplied by Bus 1H1-1.

B. DC Bus 1A-4 will be supplied by only Battery 1A as indicated by DC Bus voltage slowly trending| down over time and Vital AC Buses 1 and 1A will be supplied by Bus 1H1-2.

C. DC Bus 1A-4 will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at |125 VDC and the Vital AC Buses 1 and 1A will be supplied by 1H1-1.

D. DC Bus 1A-4 will be supplied by UPS 1A-2 as indicated by DC Bus Voltage remaining stable at |125 VBC and the Vital AC Buses 1 and 1A will be supplied by 1H1-2.

DC Bus 1A not 1A-1 is correct terminology |

Surry

References:
ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11
ND-90.3-LP-6, 125 VDC Distribution, Rev. 10
1K-A7, BATT SYSTEM 1A TROUBLE, Rev. 5
1K-A8, UPS SYSTEM TROUBLE, Rev. 1
11448-FE-1G, Sheet 1 of 1, 125V DC On6 bine Diagram - Surry Power Station Unit 1, Rev. 33

Distractor Analysis:
A. Incorrect because the battery should not be supplying the DC Bus alone. The DC Bus is being supplied by the other UPS from 1HI-2. Also, vital AC Buses 1 and 1A are being supplied by Bus 1H1-2, which is the alternate AC source.
B. Incorrect because the battery should not be supplying the DC Bus alone. The DC Bus is being supplied by the other UPS from 1H1-2.
C. Incorrect because the Vital AC Buses 1 and 1A are being supplied by Bus 1HI-2, which is the alternate AC source.
D. Correct because the other UPS will still be supplying DC Bus 1A-1 and the Alternate AC Source 1H1-2 will supply Vital AC Buses 1 and 1A.

058 boss of DC Power
AA2.01: Ability to determine end interpret the following as they apply to the loss of DC Power: That a loss of DC Bower has occurred; verification that substitute power sources have come on line.
Answer: D

42. 059A1.03 1

Which ONE of the following set of practices should be observed by operators for starting the second Main Feedwater Pump per GOP-1.5 (Unit Startup, 2% Reactor Power to Max Allowable Power) and OP-FW-004 (Main Feedwater System Operation)?

A. The second Main Feedwater Pump should be started prior to exceeding 50% power to preclude problems with main feedwater flow capability. Following pump start, if the Main Feedwater Pump Reciculation Valve is in AUTO, the operator should observe that valve closure will occur as the feed flow rises above 3000 gpm.

B. The second Main Feedwater Pump should be started between 50% power and 65% power to preclude problems with main feedwater flow capability. Following pump start, if the Main Feedwater Pump Recirculation Valve is in AUTO, the operator should observe that valve closure will occur as the feed flow rises above 3286 gpm.

C. The second Main Feedwater Pump should be started prior to exceeding 50% power to preclude problems with main feedwater flow capability. Operating the second Main Feedwater Pump on recirculation with the discharge MQV closed should be minimized to prevent overpressurization of the piping between the discharge check valve and the MOV as the system heats.

D. The second Main Feedwater Pump should be started between 50% power and 65% power to preclude problems with main feedwater flow capability. Operating the second Main Feedwater Pump on recirculation with the discharge MOV closed should be minimized *to* prevent overpressurization of the piping between the discharge check valve and the MOV as the system heats.

Surry

References:
1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power, Rev. 32
1-OP-FW-004, Main Feedwater System Operation, Rev. 8
ND-89.3-LP-3, Main Feedwater System, Rev. 12

Distractor Analysis:
A. Incorrect because recirc should modulate closed at 4000 gpm.
B. Incorrect because recirc should modulate closed at 4000 gpm.
C. Correct because of NOTE on Pg. 34 of 44 of GOP-1.5 and CAUTION on Pg 12 of 34 of OP-FW-004.
D. Incorrect because second feedwater pump should be started prior to 50% power.

059 Main Feedwater
A1.03 Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.

Answer: C

43. 059AA1.01 1

The following Unit 1 conditions exists:
- A Large Break Loss of Coolant Accident has occurred
- The "B" Train of Recirc Spray (RS), the only available train, is in service
- 1-RM-67, DISCH TNL ALERT / FAILURE, annunciates
- 1-RM-A8, RS/SW HX B ALERT/FAILURE, annunciates
- Reactor Operator notes the RSISW HX B Monitor is trending up, but the Discharge Tunnel Wad Monitor is indicating all EEEEEs with Red and Yellow Lights bit and Green Light out.

Which ONE of the following is the correct operator response?

A. Ensure no additional releases are in progress and secure RS

B. Ensure no additional releases are in progress, and increase radiation monitoring.

C. Verify all automatic actions have occurred and reset the Discharge Tunnel Digital Rate Meter and perform a source check.

D. Verify all automatic actions have occurred and raise the Discharge Tunnel Monitor set point.

Surry

References:
ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8
1-RM-G7, DISCH TNL ALERT / FAILURE, Rev. 4
1-RM-AB, RS/SW HX B ALERT/FAILURE, Rev. 3

Distraetor Analysis:
A. Incorrect because the last available RS train should not be secured, as stated in RM-G7 and RM-AB Caution Statements. Plausible because this is the correct course of action if the other train was available.
B. Correct because the last train of RS should not be secured. Other rad monitors should be checked to see if blowdowns have been diverted, to verify that there is no CCW/SW HX leak, and to verify that no CP Bld Liquid releases are occurring. Additional monitoring is called for by the ARPs due to the fact that the last train of RS should not be secured.
C. Incorrect because there are no automatic actions to verify. Plausible because the applicant may not know that there are no auto actions associated with these particular monitors. With a failed monitor, ARPs will direct a reset and source check, which adds to the plausibility.
D. Incorrect because there are no automatic actions to verify. Plausible because the applicant may not know that there are not auto actions associated with these particular monitors and it is not uncommon for an alarm setpoint to be raised to alert operators of worsening conditions. The Discharge Tunnel Monitor has the indications of being failed, therefore adjusting the setpoint is not a success path.

059 Accidental Liquid Radwaste Release
AA1.01: Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Releases: Radioactive-liquid monitor

Modified Surry ILT Bank Question #1977
Answer: B

44. 061A1.04 1

1-CN-TK-1, Emergency Condensate Storage Tank (ECST), is supplying AFW Pumps for Residual Heat Removal via Steam Generators. 1J-F4, CST 110,000 GAL LO LVL, has annunciated. ECST level is 90% and lowering.

Which ONE of the following is correct regarding refilling of the ECST?

A. Filling shall commence prior to the ECST level reaching 60,000 gallons (54%). AFW pumps must be secured prior to commencing the fill.

B. Filling may commence after the ECST bevel drops below 60,000 gallons (54%) as long as refill begins within two hours of securing the AFW pumps.

C. AFW Pumps must be secured prior to commencing the fill and the ECST must be filled within two hours.

D. Filling of the ECST shell commence prior to the ECST level reaching 60,000 gallons (54%). AFW pumps may continue to operate during the refill.

Candidates do not memorize percent values for limits for the Eme gency Condensate Storage Tanks. These limits are presented in Tech Specs in gallons. Recommend putting gallons (%) fo h choice

Surry

References:
[illegible] y [illegible] System, Rev. 19
1J- CST 110,000 GAL .O LVL, Rev. 3
ech Spec 3.6-1, Amendment N 224 and *220*

Distractor Arialysis:
A. Incorrect because AFW pumps may continue to run during refill based on AHP 1J-F4 Ncte.
B. Incorrect because volume [illegible] above 60,0C [illegible]
C. Incorrect because AFW p[illegible] do not [illegible] to be [illegible]ired for refill.
D. Correct based on all three of the above references

061 Auxiliary Feedwater
AI.04: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW source tank level.
Answer: D

45. 061AA2.01 1

If a spent fuel assembly is damaged by being dropped in the spent fuel pool, which ONE of the following pairs of radiation monitors would indicate an increase in radiation level?

A. Spent Fuel Pit Bridge Crane Radiation Monitor and Auxiliary Building Control Victoreen Area Radiation Monitor

B. Ventilation Vent Particulate Radiation Monitor and Auxiliary Building Control Victoreen Area Radiation Monitor

C. Spent Fuel Pit Bridge Crane Radiation Monitor and Ventilation Vent Gaseous Radiation Monitor

D. Ventilation Vent Gaseous Radiation Monitor and the Liquid Waste Effluent Process Monitor

Surry (Utility needs to add correct WM equipment numbers.)

References:
ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8
0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 18
0-RM-D3, 2-RM-RI-153 HIGH, Rev. 4
0-RM-B4, 1-RM-RI-152 HIGH, Rev 8

Distractor Analysis:
(maybe get some help to provide a little better distractor analysis?)

A. Incorrect because the Aux Bld Control Victoreen Area Radiation Monitor would not show an increased indication.
B. Incorrect because the Aux Bld Control Victoreen Area Radiation Monitor would not show an increased indication.
6. Correct because both monitors would show an increased indication.
D. Incorrect because a liquid waste process effluent monitor would not see the results of the failed fuel.

061 ARM System Alarms
AA2.01: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: ARM panel displays.

Surry Requal Bank Question #118
Answer: 6

46. 062A1.01 1

The following conditions were noted during the performance of 1-OPT-EG-001, Number 1 Emergency Diesei Generator Monthly Start Exercise Test:

- The EDG was loaded at a rate of 550 KW/MIN
- The Maximum load attained was 2650 KW
- The Maximum KVAR was 500 **KVAR** out
- The output voltage **was** stable at 4300 VAC

Which **ONE** of the following was in violation of the EDG Precautions and Limitations per 1-OPT-EG-001?

A. Load Rate

B. Maximum Load

C. Maximum KVAR out

D. Output voltage

Surry

References:
1-OPT-EG-001, Number 1 Emergency Diesel Generator Monthly Start Exercise Test, Rev. 24
1-OP-EG-001, Number 1 Emergency Diesel Generator, Rev. 17

Distractor Analysis:
A. Correct because the loading rate should not exceed 500 KW/MIN during normal operations.
B. Incorrect because max load rating is 2755 KW.
C. Incorrect because rnax KVAR out is 505 KVAR.
D. Incorrect because output voltage should be maintained between 4000 and 4400 VAC.

062 AC Electrical Distribution
A1.01: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significant D/G load limits.
Answer: **A**

47. 062AA1.06 1

The following Unit 1 conditions exist:
- Power = 100%
- During testing, an Intake Canal bow Level Isolation Signal is inadvertently actuated

Which ONE of the following correctly states the plant response caused by the bow Level Isolation Signal?

A. I-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will close and can only be reopened after 5 minutes.

B. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will go to 25% open and can be fully opened after 5 minutes.

C. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will close and can be reopened when the low level signal is reset.

D. 1-SW-MOV-102A and B (CCHX and SW-P-4 Supply) will go **25%** open and can be fully opened when the low level signal is reset.

Surry (Utility needs to verify technical accuracy and provide any additional reference material (electrical print?)).

References:
ND-89.5-LP-2, Service Water System, Rev. 20

Distractor Analysis:
A. Incorrect because the valves will close, but cannot be re-opened until Canal bow Level Isolation Signal is cleared. If the valves would have been closed due to a CLS, then they could have been re-opened after 5 minutes even without the CLS cleared.
B. Incorrect because the valves wilt go fully closed.
C. Correct because the valves will close, but cannot be re-opened until Canal bow bevel Isolation Signal is cleared. If the valves would have been closed due to a CLS, then they could be opened after five minutes without resetting CLS.
D. Incorrect because, as states above, the valves will close.

062 boss of Nuclear Svc Water
AA1.06: Ability to operate and / or monitor the following as they apply to the boss of Nuclear Service Water (SWS): Control of flow rates to components cooled by the SWS.
Answer: C

48. 063A4.01 1

Unit 1 was operating at 68% power when the following plant conditions developed:
- 1K-A7, BATT SYSTEM 1A TROUBLE, alarm annunciates
- "A" SG **PORV** Indicating Lights are not lit
- MSTV Indicating bights are not lit
- PORV 1455C/1456 Indicating Lights are not lit
- " A , "D", and "H" 4160 V Bus Breaker Indicating Lights are not lit
- There is no indicated letdown flow
- The Turbine Driven AFW Pump is running

Which ONE of the following describes the plant conditions assuming no other failures in addition to the cause of the above conditions?

A. The reactor will automatically trip. The turbine will automatically trip when the reactor is manually tripped.

B. The turbine will automatically trip. The reactor will automatically trip due to the automatic turbine trip

C. The reactor must be manually tripped. The turbine must also be manually tripped,

D. The reactor will automatically trip. The turbine will not automatically trip and must be manually tripped.

References:
ND-90.3-LP-6, 125Vdc Distribution, Rev. 10

Distractor Analysis:
A. Correct because the reactor will automatically trip on loss of voltage to the "A" RTB UV coil to a loss of the " A DC Bus (see ND-90.3-LP-6). The turbine will not trip until the reactor is manually tripped in accordance with E-0.
B. Incorrect because the reactor will automatically trip due to loss of voltage to the " A RTB UV coil due to the loss of the " A dc Bus.
C. Incorrect because the reactor does not need to be manually tripped to trip the reactor and the turbine will automatically trip when the reactor is tripped per E-0.
D. Incorrect because the turbine does not need to be manually tripped. The turbine will trip when the reactor is manually tripped in E-0 or when the other train of RPS occurs due to low SG levels.

063 DC Electrical Distribution
A4.01: Ability to manually operate and / or monitor in the control room: Major breakers and control power fuses.
Answer: **A**

49. 064K2.01 1

The following plant conditions exist:
- Bus 1J1 voltage drops to ~~407~~ 400 volts (80% of nominal voltage) and returns to 480 volts seven seconds later and
  remains stable
- Bus 2J1-1 voltage is ~~441~~ 445 volts(92% of nominal voltage) and stable

Which ONE of the following correctly states the source of power for Diesel Generator #3's Air Compressors?

A. Bus 1J1 remained the power supply throughout the seven second voltage drop.

B. Six seconds after the voltage dropped on Bus 1J1, Bus 2J1-1 became the power supply. Bus 2J1-1 will remain the power supply until manually transferred back to Bus 1J1

C. Six seconds after the voltage dropped on Bus 1J1, Bus 2J1-1 became the power supply. Bus 2J1-1 will remain the power supply for 30 minutes with Bus 1J1 greater than 440 volts, at which time it will automatically return to Bus 1J1.

D. Six seconds after the voltage dropped on Bus 1J1, Bus 2J1-1 became the power supply. Bus 2J1-1 will remain the power supply for six seconds with Bus 1J1 greater than 440 volts, at which time it will automatically return to Bus 1J1.

The voltages provided in the stern are much too close to the limits for the function being tested and the percentage of nominal voltage are not provided as they are in the lesson plan. Making the above indicated changes still adequately tests the KA for this question.

Surry

References:
ND-90.3-LP-I , Emergency Diesel Generator, Rev. 14
P&ID 11448-FE-1AA, Appendix R Evaluation Electrical One Line Diagram Surry Power Station Unit 1, Rev. 23
P&ID 11448-FE-1P1, 480V One Line Diagram MCC 1J1-1A Surry Power Station Unit 1, Rev. 4

Distractor Analysis:
A. Incorrect because 1J1 voltage was less than 410v for greater than 6 seconds. Therefore, 2J1-1 became the power supply after 6 seconds. The ABT will check for 2J1-1 voltage greater than 440v prior to swapping to the alternate power supply.
B. Incorrect because this is the alternate power supply and the **ABT** is a normal-seeking ABT. Therefore, at the beginning of this sequence, the power supply would have been 1J1.
C. Correct because 1J1 voltage was less than 410v for greater than 6 seconds. Therefore, 2J1-1 became the power supply after six seconds. The ABT will check for 2J1-1 voltage greater than 440v prior to swapping to the alternate power supply. When the normal power supply voltage is restored to > 440v, *a* 30 minute time delay is started. If the voltage remains above 440v for 30 minutes: then it transfers back to the normal power supply (1J1).
D. Incorrect because of the 30 minute time delay mentioned above.

064 Emergency Diesel Generator
K2.01: Knowledge of bus power supplies to the following: Air Compessors.
Answer: C

50. 065AA2.01 1

The following plant conditions exist:
- Unit 2 is in intermediate shutdown
- Operators are attempting to warm the RHR system
- An instrument air leak has developed, but the location **is** yet to be determined
- **An** Operator reports the sound of compressed air leaking in the area of the RHR pump platform.
- ~~1B-E6, IA LOW HDR PRESS/IA COMPR 1 TRBL~~ 1B-F-6, CTMT INST AIR HDR LO PRESS, has annunciated
- Instrument air pressure is approximately stable at 60 psig

Which ONE of the following correctly explains the potential effect OR warming the RHR system?

A. If the air leak is a rupture upstream of the isolation valve for the air supply to HCV-1758 (RHR Heat Exchanger Outlet Valve), the valve will fail closed. The line may be crimped if the teak will not affect vital control instruments. Operators should use the portable air bottle, via quick disconnect, to operate the valve.

B. If the air leak is a rupture upstream of the isolation valve for the air supply to HCV-1758 (RHR Heat Exchanger Outlet Valve), the valve will fail open. The line may be crimped if the leak will not affect vital control instruments. Operators should use the portable air bottle, via quick disconnect, to operate the valve.

C. If the air leak is a rupture upstream of the isolation valve for the air supply to HCV-1142 (CVCS Flow Regulator Control Valve), the valve will fail closed. The line may be crimped if the leak will not affect vital control instruments.

D. If the air leak is a rupture upstream of the isolation valve for the air supply tu HCV-1142 (CVCS Flow Regulator Control Valve), the valve will fail open. The line may be crimped if the leak will not affect vital control instruments.

Sur9

References:
ND-88.2-LP-1, Residual Heat Removal System Description, Rev. 8 (Pages 9, 10, 11)
ND-88.2-LP-2, Operation of Residual Heat Removal System, Rev **15**
P&ID **11448-FM-087A** Sh **2** of 2, Residual Heat Removal System, Rev. 26
P&ID **11448-FM-075E** Sh 1 of 2, Compressed Air System, Rev. 43
1B-E6, IA LOW HDR PRESS/IA COMPR 1 TRBL, Rev. 9

Bistractor Analysis:
A. Incorrect because HCV-1758 fails open and cannot be operated with a portable air bottle. Plausible because the applicant may get consfused on which valve in this flowpath has the portable air bottle feature.
B. incorrect because HCV-1758 cannot be operated with a portable air bottle. Plausible because the applicant may get consfused on which valve in this flowpath has the portable air bottle feature.
C. Correct because HCV-1142 **is** fail closed and this **is** the flow path for system warmup. **ARP** states that leaks may be stopped via crimping if the leak will not affect vital instrumentation.
D. Incorrect because HCV-1142 fails closed. Plausible because the applicant may get confused OR failure modes of HCV-1142, especially since it does have a backup air bottle feature for App. R purposes.

065 **Loss** of Instrument Air
AA2.01: Ability to determine and interpret the following as they apply to the **loss** of instrument air:
Cause and effect of low pressure instrument air alarm.
Answer: C

51. 067G2.4.18 1

In FCA-8.00, Limiting Auxiliary Building Fire, if Charging Pump CC Pumps are not running, the operator is directed to shift charging pump suction to the RWST. Which ONE of the following describes the basis for this step?

A. Suction is shifted to the RWST to maximize boron injection before the charging pumps overheat and are lust due to a time-overcurrent breaker trip.

B. Suction is shifted to the RWST to maximize boron injection before the charging pumps overheat and are lost due to a instantaneous-overcurrent breaker trip.

C. The loss of Charging Pump CC will eventually result in a loss of VCT level due to a lass of makeup; therefore suction is shifted to the RWST.

D. The RWST supplies cooler water to the Charging Pumps; thereby minimizing the cooling requirements for the Charging Pumps.

Surry (Utility needs tu verify technical acuracy and supply additional supporting material if any is availble.)

Refernces:
ND-95.6-LP-3, Safe Shutdown Fire FCAs, Rev. 5
0-FCA-8.00, Limiting Auxiliary Building Fire, Rev. 13

Distractor Analysis:
A. Insurrect because the concern is with overheating the pump, not maximizing boron injection prior to the pump overheating. Supplying cooler RWST water will reduce the pump temperatures.
B. Incorrect because the concern is with overheating the pump, not maximizing boron injection prior to the pump overheating. Supplying cooler RWST water will reduce the pump temperatures.
C. Incorrect because VCT level will not be reduced as a result of no CC.
D. Correct because cooler RWST water will help reduce pump temps when CC is lost.

067 Plant Fire On-Site
G2.4.18: Knowledge of specific bases for EOPs.
Answer: D

52. 068K4.01 1

The following Conditions exist:
- Both Units are at 100% Power
- Unit 1 Operators have discovered indication of a small tube leak in the "A" Steam Generator for their Unit
- Spent Fuel is being moved in the Spent Fuel Storage Pool to facilitate rack inspections
- 0-RM-M4, 1-VG-RI-104 HIGH, alarms
- All Radiation Monitors appear to be operating satisfactorily
- Ventilation and Radiation Monitors are in their normal alignment

Which ONE of the following could cause WM-VG-104 (#1 Vent Stack RM) to detect higher than normal activity?

A. A Steam Generator Tube Leak on Unit 1.

B. A spill of high activity coolant in the Chemistry Hot Lab.

C. A spill of high activity coolant in the High Rad Sample System Room.

D. A dropped fuel assembly in the fuel building.

Surry

References:
0-RM-M4, 1-VG-81-104 HIGH, Rev. 2
0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 18
ND-95.3-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8

Distractor Analysis:
A. Incorrect because the normal configuration for the ventilation system would not have Main Condenser Air Ejector aligned to discharge to the Number 1 Vent Stack upstream of Radiation Monitor 1-VG-RM-104.
B. Correct because 0-RM-M4 alarming could be caused by a coolant spill in the Chem Hot Lab according to the ARP.
C. Incorrect a spill in the High Radiation Sample System Room would not cause this alarm according to the ARP.
D. incorrect because fuel clad damage would not be detected by RM-VG-104 when in its normal configuration. **0-AP-22.00** does not list RM-VG-104 as a potential means of indication for damaged fuel dad.

068 Liquid Radwaste
K4.01: Knowledge of design feature(s) and / or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids.

Surry Requal Exam Bank Question #462 (ID:ARP0076)

Answer: B
Utility needs to help verify this (verify all distractors are wrong) and provide any additional references. For example, ensure that there is only one expected "normal aligment", such that a tube leak wiil not **be** expected to yield this alarm.

53. 071K4.06 1

A discharge of a waste gas decay tank is in progress when RM-GW-101 reaches the high alarm setpoint and alarm 0-RM-K3, 1-GW-RI-101 HIGH, annunciates. Which ONE of the following is NOT an automatic action initiated by the high radiation levels from the waste gas decay tank release?

A. 1-GW-FCV-101, Decay Tank Bleed Isolation Valve, closes.

B. 1-GW-FCV-160, CTMT Vacuum Pump Discharge Isolation Valve closes.

C. 1-GW-FCV-260, CTMT Vacuum Pump Discharge Isolation Valve, closes.

D. Associated vacuum pumps trip.

Surry (Utility needs to verify technical accuracy)

References:
ND-92.4-LP-1, Gaseous and Liquid Waste Processing Systems, Rev. 8
ND-93.5-LP-1, Pre-TMI Radiation Monitoring System, Rev. 8
0-RM-K3, I-GW-RI-101 HIGH, Rev. 0

Distractor Analysis:
A. Incorrect because according to ARP, this valve will close on reaching the high alarm setpoint.
B. Incorrect because according to ARP, this valve will close on reaching the high alarm setpoint.
C. Incorrect because acording to **ARP,** this valve will close on reaching the high alarm setpoint.
D. Correct because the pumps must be manually secured if GW-160 or GW-260 are closed. This info is in a CAUTION in the ARP and a step is provided in the ARP to secure the pumps foliowing the closure of GW-160 / 260.

071 Gaseous and Liquid Waste Processing Systems
K4.06: Knowledge of design(s) features and / or interlocks which provide for the following: Sampling and monitoring of waste gas release tanks.
Answer: D

54. 073G2.1.23 1

Which QNE of the following is sufficient ~~conclusive~~ CONCLUSIVE indication of RCS leakage and not some other event to warrant a correct entry into AP-16.00, Excessive RCS Leakage?

A. Wising containment humidity, rising containment temperature, and rising containment pressure.

B. Rising steam generator water level, rising charging flow, and rising Condenser Air Ejector Radiation Monitor reading.

C. Rising Condenser Air Ejector Radiation Monitor reading, rising steam generator blowdown radiation monitor reading, and stable containment pressure.

D. Rising containment sump level, lowering pressurizer pressure, and rising containment pressure.

Each of these would be entry criteria for &I- 16 as indicated on the Entry Conditions on *the* cover page of AP-16. The additions to stem need to be added to clarify exactly what you are asking for in the question.

Surry

References:
1-AP-16.00, Excessive RCS Leakage, Rev. 11
1-AP-24.00, Minor SG Tube beak, Rev. 8
I-E-0, Reactor Trip or Safety Injection, Rev. 46
ND-93.5-LQ-1, Pre-TMI Radiation Monitoring System, Rev. 8

Distractor Analysis:
A. Incorrect because a steam line break can cause containment humidity, temperature, and pressure to rise. Distractor is plausible because these are all possible for an RCS leak.
B. Correct because SG water level is indication that there may be a tube leak or steam/feed mismatch. The charging flow and Air Ejector Rad monitor corroborates that the problem is tube leakage.
C. Incorrect because these parameters are indications that there may be a tube leak; however, these same indications may present themselves with a fuel failure or crud burst. Distractor is plausible because these parameters may indicate as stated if RCS leakage actually exists. Distractor is incorrect because these parameter trends may be caused by increased RCS activity
D. Incorrect because the combination of these parameters may be caused by a steam leak/break. Distractor is plausible because these parameters may actually change as indicated during an RCS leak.

073 Process Radiation Monitoring
62.1.23: Ability to perform specific system and integrated plant procedures during all modes of plant operation.
Answer: B

55. 076AK2.01 1

Hydrogen peroxide has just been added to Unit 2 WCS resulting in an increase in the primary coolant activity. The first indication that the activity level has increased will be seen on the __________ and the team should __________

A. Containment particulate radiation monitor; increase flow through the letdown cation bed.

B. Letdown radiation monitor; monitor letdown filter differential pressure.

C. Letdown radiation monitor; monitor seal return filter differential pressure.

D. Containment particulate radiation monitor; decrease flow through the letdown cation bed.

References:
ND-93.05-LP-1, Pre-TMI Radiation Monitoring System
ND-88.3-LP-3, Seal Injection, Rev. 6

Distractor Analysis:
A. incorrect because containment particulate radiation monitor would not change significantly.
B. Correct because letdown radiation monitors would indicate quickly due to hydrogen peroxide increasing reactor coolant activity and letdown filter dP would also rise.
C. Incorrect because the hydrogen peruxide should not affect the seal return dP, at least not as readily or as soon as the letdown filter dP. There is 8 gal of CVCS water that goes to each RCP for seal injection. Five of these gallons flows down the shaft past the thermal barrier and ends up in the RCS. The other three gallons eventually passes through the seal return filter. The CVCS water that enters the RCP seal area has already been filtered prior to getting to the RCP seals. This prefiltering is designed to protect the seals. The water corning from the RCP seal area should be relatively clean CVCS water, not RCS water; therefore making the seal return filter a relatively poor indicator of a crud burst.
D. Incorrect because containment particulate radiation monitor would not change significantly.

Surry Bank ILT Exam Question #1606

076 High Reactor Coolant
AK2.01: Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors.
Answer: B

56. 076K2.04 1

The following Unit 1 conditions exist:
- A Large Break LQCA occurred **45** minutes ago
- Recirculation Spray is operating
- MCC 1H1-2 de-energizes

Which ONE of the following correctly describes the impact on Service Water to and from the Recirc Spray Heat Exchangers?

A. Recirc Spray Heat Exchanger 1-RS-E-1A Service Water Inlet (MOV-SW-104A) and Outlet (MOV-SW-105A) Valves de-energize.

B. Recirc Spray Heat Exchanger I-RS-E-1B Service Water Inlet (MOV-SW-104B) and Outlet (MOV-SW-105B) Valves de-energize.

C. Recirc Spray Heat Exchanger Service Water Inlet (MOV-SW-103A) and Recirc Spray Heat Exchanger 1-RS-E-1A Service Water Inlet (MOV-SW-104A) Valves de-energize.

D. Recirc Spray Heat Exchanger Service Water Inlet (MOV-SW-103B) and Recirc Spray Heat Exchanger 1-RS-E-1B Service Water Inlet (MOV-SW-104B) Valves de-energize.

References:
ND-91-LP-6, Recirculation Spray System, Rev. 9
ND-89.5-LP-2, Service Water System, Rev. 20
P&ID 11448-FE-1M, Sh 1 of 1, 480V One Line Diagram Surry Power Station - Unit 1, Rev. 59
P&ID 11448-FE-1L, Sh 1 of 1, 480V One Line Diagram Surry Power Station - Unit 1 ,Rev. **52**

Distractor Analysis:
A. Correct because MOV-SW-104A and 105A are both powered from 1H1-2.
B. Incorrect because MOV-SW-104B and 105B are both powered from 1J1-2.
C. Incorrect because MOV-SW-103A is powered from 1H1-1.
D. Incorrect because MOV-SW-103B is powered from 1J1-1 and MOV-SW-104B is powered from 1J1-2.

076 Service Water
K2.04: Knowledge of bus power supplies to the fullowing: Reactor building closed cooling water.

Answer: A

Memorization to this level of detail is not required knowledge. Request replacing with the following:

The following Unit 1 conditions exist:

- A Large Break LOCA occurred **45** minutes ago
- Recirculation spray is operating
- 4160 V "H" bus de-energizes

Which one of the following correctly describes the impact on Service Water to and from the Recirc Spray Heat Exchangers?

a. The Service Water Outlet Valves (MOV-SW-105s) from each heat exchanger de-energize

b. The Service Water Inlet Valves (MOV-SW-104s) from each heat exchanger de-energize

c. The Service Water Inlet Valves (MOV-SW-104s) and Service Water Outlet Valves (MOV-SW-105s) from both Inside Recirc Spray Heat Exchanger de-energize.

d. The Service Water Inlet Valves (MOV-SW-104s) and Service Water Outlet Valves (MOV-SW-105s) from one Inside and one Outside Recirc Spray Heat Exchanger de-energize.

57. 078A4.01 1

Unit 1 is at 50% power and the team is experiencing problems controlling feedwater flow. An instrument Air **Low** Pressure Alarm is received in the Control Room. While monitoring Instrument Air pressure. the RO notes pressure is 50 psig and slowly lowering.

Which ONE of the following actions should be taken?

A. Commence a slow power reduction to Hot Shutdown.

B. Commence a fast power reduction to Cold Shutdown.

C. Trip the Reactor and go to I-E-0, Reactor Trip or Safety Injection.

D. Isolate Service Air from Instrument Air and start the Sullair Diesel

Surry

References:
ND-92.1-LP-1, Station Air Systems, Rev. 13
1B-E6, IA LOW HDR PRESS / IA COMPR 1 TRBL, Rev. 9
0-AP-40.00, Mon-recoverable boss of Instrument Air, Rev. 17

Distractor Analysis:
A. Incorrect because 1B-E6 and AP-40.00 directs rx trip, not power reduction.
B. Incorrect because 1B-E6 and AQ-40.00 directs rx trip, not power reduction. (Initial distractor *from* exam bank was changed because it may have been a second correct answer).
C. Correct because this **is** the guidance provided by 1B-E6 and AP-40.00.
D. Incorrect because 1B-E6 and AP-40.00 directs rx trip, not power reduction when pressure reaches 50 psig.

078 instrument Air
A4.01: Ability to manually operate and / **or** monitor in the control room: Pressure gauges.

Surry Requal Exam Bank Question 428
Answer: C

58. 078K4.02 1

With ALL air systems aligned in the automatic mode, which ONE of the following describes the operation of the Station Instrument Air (IA) System for Unit 1?
(Assume no operator action is taken.)

A. Instrument Air is normally supplied by the Service Ais System and the system is backed up by IA when IA pressure reaches 95 psig.

B. Instrument Air is normally supplied by IA Compressors and the system is manually backed up by the Sullair Diesel.

C. Instrument Air is normally supplied by the Sewice Air System and is backed up by the IA System when IA pressure reaches 90 psig.

D. Instrument Air is normally supplied by the Service Air System and is backed up by the Condensate Polishing Instrument Air System when IA pressure reaches 98 psig.

Surry

References
ND-92.1-LP-1, Station Air Systems, Rev. 73

Distractor Analysis:
A. incorrect because pressure must drop below 90 psig for IA to backup Service Air.
B. Incorrect because the IA System is normally supplied by Service Air.
C. Correct because Service Air is the normal supply and IA is the backup when pressure drops below 90 psig.
D. Incorrect because IA is not backed up by the Condensate Polishing Instrument Air System when pressure drops *to* 98 psig. It is backed up by the IA System when pressure drops below 90 psig.

078 Instrument Air
K4.02: Knowledge of the IAS design feature(s) and or interlock(s) which provide for the following: Cross-over to other air systems.

Surry Requal Bank Question #512

Answer: C

59. 103A4.04 **1**

The following Unit 4 conditions exists:
- A steam line rupture in Containment occurred several minutes ago
- Maximum Containment Pressure reached 24 psia
- Containment Pressure Transmitters now read:
  - PT-LM-100A = 17.7 psia
  - PT-LM-100B = 17.8 psia
  - PT-LM-100C = 17.6 psia
  - PT-LM-100D = 17.9 psia

Which ONE of the following correctly describes resetting of Consequence Limiting Safeguards (CLS) given the above conditions?

A. The CLS TRAIN A(B) RESET PERMISSIVE annunciator is lit. CLS HI and CLS HI-HI may be reset at this time. Upon reset, the Hi CLS relays will en[redacted] the Hi-Hi CLS relays will de-energize..

B. Neither CLS HI or CLS HI-HI may be reset at this time. The Hi CLS relays are de-energized and the Hi-Hi CLS relays are energized.

C. The CLS HI-HI RESET PERMISSIVE annunciator is lit. CLS HI-HI may be reset at this time. Upon reset the Hi-Hi CLS relays will de-energize.

D. Neither CLS HI or CLS HI-HI may be reset at this time. The ~~multiplying~~ Hi CLS relays are energized and the Hi-Hi CLS relays are de-energized.

Justification - The Hi CLS relays energize when reset and the Hi-Hi CLS relays de-energize when reset.

Surry

References:
ND-88.4-LP-2, Containment Vessel, Rev. 8
MD-91-LP-5, Containment Spray System, Rev. 13

Bistractor Analysis:
A. Incorrect because pressure must be reduced to less than 14.2 psia on 2/4 channels to reset both Hi and Hi-Hi subsystems.
B. Correct because pressure must be reduced to less than **44.2** psia on 2/4 channels to reset both Hi and Hi-Hi subsystems. Also, when CLS is actuated, the multiplying relays are de-energized.
C. Incorrect because pressure must be reduced to less than 14.2 psia on 2/4 channels to reset both Hi and Hi-Hi subsystems. Also, when CLS is actuated, the multiplying relays are de-energized.
D. Incorrect because when CLS is actuated, the multiplying relays are de-energized.

103 Containment
A4.04: Ability to manually operate and / or monitor in the control room: Phase A and Phase B resets.

Answer: B

60. G2.1.11 1

The following Unit 1 conditions existed:
- Plant ~~is~~ was at 74% power after just completing a rapid power reduction due to High Pressure Heater Drain Pump problems
- Axial Flux Difference was outside of the Parget Band on 11/03/2003 from 0800 hours to 0845 hours
- Axial Flux Difference was outside of the Target Bend on 11/04/2003 from 0740 hours to 0840 hours
- The Axial Flux Difference has remained within the Technical Specification Limiits of Figure 3.12-3, Axial Flux Difference Limits As A Function Of Rated Power, for the entire time

Which ONE of the following actions are required by Technical Specifications?

A. Reactor power was required to be less than 50% by 0825 hours on 11/04/2003.

B. Reactor power was required to be less than 50% by 0855 hours on 11/04/2003.

C. Reactor power was required to be less than 50% by 0910 hours on 11/04/2003.

D. No power reduction was required, but power should not have been raised above 75% until Axial Flux Difference was within the Target Bend.

Surry

Reference:
Technical Specification 3.12.B.4.b.(1), Amendment No. 186
Technical Specification 3.12.B.4.b.(2), Amendment No. 186

Bistractor Analysis:
A. Correct because AFD may deviate from its target band for one hour within a 24 hour period. When this is violated, then power must be reduced to less than 50% within 30 minutes. From 11/03 @ 0800 hrs to 11/04@ 0755 hrs a total of one hour outside of target band was accumulated. Therefore, by 0825 hrs (30 minutes later) power must be less than 50%.
B. Incorrect because ~~because~~ the correct answer is as described in above analysis. Plausible because 0855 hours is 60 minutes after 0755 hrs, which is when the 30 minute clock starts to have power less than 50%.
C. Incorrect because the correct answer is as described in above analysis. Plausible because 0910 hrs is 30 minutes after 0840 hrs, which was given as the second time frame where AFD was outside of its target band.
D. Incorrect because ~~because~~ the correct answer is as described in above analysis. Plausible because candidate may confuse 50% and 75% power restrictions.

62.1.11
Knowledge of less than 1 hour technical specification action statements for systems.
Answer: A

61. G2.1.25 1

The following conditions exist:
- Unit 1 has been shutdown for 10 days for SG tube plugging
- RCS water level is being maintained at 12.4 feet as indicated on 1-RC-LI-100A
- The "B" and "C" loops are isolated with the primary and secondary SG manways removed for SG tube plugging
- The reactor vessel head is tensioned
- The "A" RHR pump is in operation with oscillating amperage indications
- Flow indication 1-RH-FI-1605 is oscillating between 2500 and 2700 gpm.

Which ONE of the following actions is appropriate for the SRO to direct in accordance with AP-27.00, boss of Decay Heat Removal Capability?
(AP-27.00 Attachments 1 and 2 provided)

A. Raise RCS level to 12.5 feet as indicated on 1-RC-LI-100A and stabilize flow at 2600 gpm.

B. Throttle open 1-RH-HCV-1758 and throttle close 1-RH-FCV-1605 to reduce RHR flow to 2200 gprn.

C. Throttle close 1-RH-HCV-1758 and throttle open 1-RH-FCV-1605 to reduce RHR flow to 1200 gpm.

D. Throttle close 1-RH-FCV-1605 to reduce RHR flow to 2200 gpm and raise level to 12.5 feet as indicated on 1-RC-LI-100A.

Surry

References:
1-AP-27.00, Loss of Decay Heat Removal Capability, Rev. 10
ND-88.2-LP-1, Residual Heat Removal System Description, Rev. 8
ND-88.2-LP-02, Operation of Residual Heat Removal System, Rev. 15
MD-95.2-LP-12, Loss of RHR Events, Rev. 9

Distractor Analysis:
A. Incorrect because AP-27 Att. 2 indicates that 12.5 feet is in the unacceptable region of operation fer 2600 gpm RHR flow rate.
B. Incorrect because AP-27 Att. 2 indicates that 2200 gpm RHR flow rate is in the unacceptable region of operation for 12.4 feet.
C. Incorrect because AP-27 Att. 1 indicates that 1200 gprn RHR flow rate is less than the required flow rate of 2200 gpm.
D. Correct because these actions place the plant in an acceptable region of AP-27 Att. 1 and 2 for required flow rate for 10 **days** after shutdown.

AP-27 Att. 1 and 2 will need to be provided to the applicant.

G2.1.25: Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.
Answer: D

62. **G2.2.221**

Which ONE of the following is correct with respect to Technical Specifications?

A. The Safety Limit for core thermal power is 109% of Rated Thermal Power and the WCS pressure limit is 2735 psig.

B. The Safety Limit for core thermal power is 109% of Rated Thermal Power and the single loop loss of flow reactor trip shall be unblocked when power range nuclear flux is greater than or equal to 50% of Rated Thermal Power.

C. The reactor trip on low pressurizer pressure, high pressurizer level, turbine trip, and low reactor coolant flow for two or more loops shall be unblocked when pewer is greater than **or** equal to 10% ~~of~~ of Rated Thermal Power.

D. The source range high flux, high setpoint trip shall be unblocked when the intermediate range nuclear flux is less than or equal to $5x10^{-10}$ amperes.

Sur9

References:
Technical Specification 2.1 (Amendments 116); 2.2 (Amendments 203); 2.3 (Amendments 175, 176,206)

Distractor Analysis:
A. Incorrect because the safety limit for core thermal power is 118%.
B. Incorrect because the safety limit for core thermal power is 118%.
C. Correct because this is the correct statement taken from Tech Specs.
D. Incorrect because source range high flux. high setpoint trip shall be unblocked when the intermediate range nuclear flux is less than or equal to $5x10^{-11}$ amperes.

Generic K/A 2.2.22
Knowledge of limiting conditions for operations and safety limits.
Answer: C

63. **G2.2.27** 1

Which ONE of the following correctly states the level of authorization needed for bypassing the Manipulator Crane Overload Interlock?

A. Refueling SRO or Fuel Handling Supervisor

B. Refueling SRO and Shift Supervisor

C. SNSOC and Refueling SRO

D. SNSOC only

Surry

References:
VPAP-1401, Conduct of Operations, Rev. 11 (Section 6.5)

Distractor Analysis:
A. Incorrect because SNSOC pre-approval is needed per 1-QP-FH-015 Step 4.12.
B. Incorrect because **SNSOC** pre-approval is needed per 1-OP-FH-015 Step 4.12.
C. Correct because SRO approval is needed per I-OP-FH-015 Step 4.10 AND SNSOC pre-approval is needed per 1-OP-FH-015 Step 4.42.
D. Incorrect because SRO approval is needed per 1-OP-FH-015 Step 4.18.

G2.2.27
Knowledge of the refueling process.
Answer: C

64. G2.3.10 1

The following conditions exists:
- Unit 2 is at full power
- Unit 1 is in refueling
- Fuel repair is being performed
- A damaged fuel rod is raised too close to the surface of the water
- Area radiation monitors alarm in the vicinity of the fuel movements
- Operators enter 0-AP-22.00, Fuel Handling Abnormal Conditions
- All components operate as designed

Which ONE of the following are immediate actions of AP-22.00?

A. Stop fuel handling operations, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C end 1-VS-MOD-103D, Dump Cable Vault Air Bottles by closing 1-VS-MOD-2038.

B. Stop fuel handling operations, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C and 1-VS-MOB-1038, Bump MER 3 Air Bottles by closing 1-VS-MOD-103A.

C. Evacuate the affected areas, Secure Normal MCR Ventilation by closing 1-VS-MOD-103C and 4-VS-MOD-103D, Dump MER 3 Air Bottles by closing 1-VS-MOD-103A.

D. Stop fuel handling operations, Evacuate the affected areas, Stop Main Control Room Fans 1-VS-F-15 and 4-VS-AC-4.

Surry

References:
0-AP-22.00, Fuel Handling Abnormal Conditions, Rev. 48

Distractor Analysis:
A. Correct these are all listed as immediate actions of AP-22.00.
B. Incorrect because 1-VS-MOD-103A is in the RNO column to be performed if 1038 does not close. However, the stem states that all equipment operates as designed, so the operator would not go to the RNO column.
C. Incorrect because 1-VS-MOD-103A is in the RNO column to be performed if 1038 does not close.
D. Incorrect because stopping MCR Ventilation Fans is not an immediate action.

G2.3.10: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.
Answer: A

65. G2.3.2 1

Work in a radiation area must be performed. The following conditions exist:
- A point source is present and emits 50 mrem/hour at 1 foot
- The air has a Derived Air Concentration (DAC) of 10

Which ONE of the following methods will result in the lowest amount of accumulated dose?

A. Two workers using hand tools can perform the work in one hour at a distance of two feet wearing no respirator.

B. Three workers using remote tools perform the work in two hours at a distance of six feet wearing no respirator.

C. Two workers using hand tools perform the work in four hours at a distance of two feet wearing a respirator with a protection factor of 50.

D. Three workers using remote tools perform the work in 10 hours at a distance of six feet wearing a respirator with a protection factor of 50.

BAC is not something operators deal with We need to add 1 DAC = 2 5 mr/hr to the equation sheet.

Surry

References:
Dominion Nuclear Employee Training Manual Volume II BRWT, RPT, CSET, SCAT, FWT, Rev. 11, January, 2003.

Distractor Analysis:

A. Incorrect: 75 mrem > 56.7 mrem. {[(2 men)(hr)(50mrem/hr)$(1/2)^2$]+[(10 DAC) (2 men)(hr)(2.5 mrem/DAC-HR)] = 75 mrem}

B. Incorrect: 158.3 mrem > 56.7 mrem. {[(3 men)(2 hr)(50mrem/hr)$(1/6)^2$]+[(10 DAC) (3 men)(2 hr)(2.5 mrem/DAC-HR)] = 158.3 mrem)

C. Incorrect: 104 mrem > 56.7 mrem. {[(2 men)(4 hr)(50mrem/hr)$(1/6)^2$]+[(10 DAC) (1/50)(2 men)(4 hr)(2.5 mrem/DAC-HR)] = 104 mrem}

D. Correct: [(3 men)(10 hrs)(50 mrem/hr)$(1/6)^2$] + [(10 DAC)(1/50)(3 men)(10 hrs)(2.5 mrem/1 DAC-HR)] = 41.7 + 15 = 56.7 mrem.

G2.3.2
Knowledge of facility ALARA program.

Answer: D

66. G2.3.9 1

The following Unit 1 conditions exist:
- The RCS temperature is 190 °F.
- Operators are performing Section *5.2* of 1-OP-VS-001, Containment Ventilation, to place the Containment Purge System in service using 1-VS-F-58A or 1-VS-F-58B, Filter Exhaust Fans.
- The Containment Purge Form requires ~~5000~~ 10,000 cfm purge flow.

Justification – Memorizing the flow limit of 3,000 cfm for VS-MOD-101 is not required knowledge. Understanding methodology for obtaining low flow rates and high Row rates during purge operation is required knowledge. Recommend changing value in stem of question *to* 10.000 cfm. This provides a greater buffer from 3,000 cfm but still allows testing purge methodology knowledge.

Which ONE of the following correctly states selection criteria, in accordance with 1-OP-VS-002, for choosing which valve to use for obtaining the correct purge flow rate?

A. 1-VS-MOV-100D (Ctmt Purge Exh) should be throttled instead of 1-VS-MOV-101 (Ctmt Purge B/P) due to the high flow rate required by **the** Containment Purge Form.

B. 1-VS-MOV-101 (Ctmt Purge B/P) should be throttled instead of 1-VS-MOV-100D (Ctmt Purge Exh). This is due to the need to open the supply breaker to 1-VS-MOV-100D in order to throttle it. Opening the breaker will prevent automatic CTMT Purge isolation.

C. 1-VS-MOV-101 (Ctrnt Purge B/P) should be throttled instead of 1-VS-MOV-100D (Ctmt Purge Exh) due tu the low flow rate required by the Containment Purge Form.

D. 1-VS-MOV-100D (Ctmt Purge Exh) should be throttled instead of 1-VS-MOV-101 (Ctmt Purge B/P). This is due to the need to open *the* supply breaker to 1-VS-MOV-101 in order *to* throttle it. Opening the breaker will prevent automatic CTMT Purge isolation.

Surry

References:
1-OP-VS-001, Containment Ventilation, Rev. 20

Distractor Analysis:
A. Correct because 100D should be throttled due to the Containment Purge Form allowing more than 3000 cfm. The bypass will not have enough capacity at this flow rate.
B. Incorrect because even though auto containment purge isolation will not occur with the breaker open, the procedure still directs the use of 100D due to the high flow rate. Plausible because applicant may think it logical to not intentionally incapacitate auto containment isolation.
C. Incorrect because with the flow rate greater than 3000 gpm, 1008 should be used. Plausible because 3000 gpm is not a very high flow rate.
D. Incorrect because the bkr does not need to be opened and at 5000 gpm, the procedure directs 101 to be used for fine tuning the flow rate. Plausible because preventing auto ctmt purge isolation is a concern when using 100D.

G2.3.9: Knowledge of the process for performing a containment purge.
Answer: **A**

67. G2.4.11 1

Which ONE of the following correctly states the requirements for performing immediate action steps within emergency procedures?

A. Immediate action steps must be performed in the order in which they appear in any procedure.

B. Immediate action steps may be performed in any order, except for the first four immediate action steps of E-0, Reactor Trip or Safety Injection, which must be performed in the order in which they appear in the procedure.

C. Immediate action steps may be performed in any order except for the first four immediate action steps of E-0, Reactor Trip or Safety Injection, and the immediate action steps of FR-S.1, Response to Nuclear Generation / ATWS, which must be performed in the order in which they appear in the procedure.

D. Immediate action steps may be performed in any order except for the immediate action steps of FR-S.1, Response to Nuclear Generation / ATWS, and ECA-0.0, Loss of All AC Power, which must be performed in the order in which they appear in the procedure.

Surry

References:
ND-95.3-LP-2, Emergency Procedure Writer's Format, Rev. 8
(Have Utility add any addional references that may support answer.)

Distractor Analysis:
A. Incorrect because only immediate actions of E-0 and FR-S.I must be performed in the order in which they appear in the procedure.
B. Incorrect because only immediate actions of E-0 and FR-S.I must be performed in the order in which they appear in the procedure.
C. Correct because immediate actions of E-0 and FR-S.1 must be performed in the order in which they appear in the procedure. This requirement / expectation is stated in ND-95.3-LP-2 Page 12.
D. Incorrect because ECA-0.0 are not required to be performed in any specific order.

G2.4.11: Knowledge of abnormal condition procedures
Answer: C

68. G2.4.12 1

A situation presents itself that requires a Reactor Operator (RO) to take quick decisive action to ensure Station Safety. Personnel are net in immediate danger and the action requires no reactivity manipulations.

Which ONE of the following correctly describes the requirements for performing the actions?

A. The RO may take necessary action without prior approval from another licensed operator.

B. The RO must immediately request approval from the Unit SRO to perform the action and only take action after approval is granted.

C. The RO may take action only after another licensed operator has been notified and concurs with the action.

D. The RO may take action only after obtaining a peer check to concur with the action.

Surry

References:
OPAP-0006, Shifl Operating Practices, Rev. **4**

Distractor Analysis:
A. Correct because OPAP-0006 Step **6.1**0.3 states, "During emergencies, Shift Team members may take necessary immediate actions required to ensure personnel and Station safety without prior approval. The Shift Supervisor shall be promptly informed of these actions."
B. Incorrect because action may be taken prior to obtaining permission.
C. Incorrect because action may be taken prior to notifying or obtaining permission from another Team Member.
D. Incorrect because immediate action is authorized to protect the Station.

G2.4.12: Knowledge of general operating crew responsibilities during emergency operations.
Answer: A

69. **G2.4.49** 2

Given the following conditions:
- Reactor Power = 85%
- Control Rods are in automatic
- Control Bank D begins tu insert without a turbine runback
- Tave and Tref are matched within 0.5 °F

Which ONE of the following describes the correct immediate operator response to these conditions?

A. Verify quadrant power tilt and axial flux difference within limits.

B. Place ROD CONT MODE SEL switch in MANUAL.

C. Manually trip the reactor.

D. Verify IRPI operating properly.

Surry

References:
0-AP-1.00, Rod Control System Malfunction, Rev. 9.

Distractor Analysis:
A. Incorrect because the initial response *is* to place ROD CONT MODE SEL switch in MANUAL.
B. Correct per AP-1.00.
C. Incorrect because this would not be performed until ROD CONT MODE SEL switch was placed to MANUAL and rad motion had stopped.
D. Incorrect because AP-I.00 directs placing ROD CONT MODE SEL switch in MANUAL as an immediate action.

G2.4.49
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
Answer: B

70. WE04EK3.2 1

Which ONE of the following correctly states actions contained in 1-ECA-1.2, LOCA Outside Containment, and the reasons for those actions?

A. Open 1-SI-MOV-1890A (LHSI to Hot Leg) or 1-SI-MOV-1890B (LHSI to Hot Leg) to provide a flow path for bow Head Safety Injection. Then close 1-SI-MOV-1890C (LHSI to Cold Legs) and monitor RCS pressure.

B. if closing 1-SI-MOV-1890C (LHSI to Cold Legs) does not result in an RCS pressure rise then allow it *to* remain closed because this will give operators time to check Aux Building alarms while the flow path is isolated.

C. If the leak ***is*** not identified and isolated then transition to 1-E-1, Loss of Reactor or Secondary Coolant, because RCS inventory is continued to be lost outside of containment.

D. If closing 1-SI-MOV-1890C (LHSI to Cold Legs) results in an RCS pressure rise, then place the LHSI pumps in PTL because their suction valves from the RWST will be closed to isolate potential leak paths.

References:
ND-95.3-LP-21, ECA-1.2 LOCA Outside Containment, Rev. 7
ECA-1.2, LOCA Outside Containment, Rev. 5

Bistractor Analysis:
A. Incorrect because ECA-1.2 does not give any direction to open 1-SI-MOV-1890A & B. These valves should be left in the closed position. This distractor is plausible because ECA-1.2 does give guidance to close 1890C.
B. Incorrect because if 1-SI-MOV-1890C is closed and RCS pressure is still decreasing, then the leak was not isolated and the valve needs to be re-opened. This is the normal SI flow path and it is important to re-establish this path if closing the valve did not isolate the leak.
C. Incorrect because if the leak is not isolated, then the correct transition wouid be to go to 1-ECA-1.1, Boss of Emergency Coolant Recirculation.
D. Correct because if RCS pressure rises upon closure of 1-SI-MOV-1890C, then the leak was isolated and 1-ECA-1.2 directs the LHSI pumps to be placed in PTL and the suction valves from the RWST to be closed.

WE04
EK3.2: Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment): Normal, abnormal, and emergency operating procedures associated with (LOCA Outside Containment).
Answer: D

71. WE06EK3.1 1

1-FR-C.1, Response to Inadequate Core Cooling, is being performed. Which ONE of the following is the reason RCPs are stopped prior to depressurizing the SGs to less than 150 psig during an inadequate core cooling event?

A. RCP operation with the SGs at atmospheric pressure is prohibited due to excessive hydraulic stress on the SG U-tubes.

B. The SGs will depressurize more quickly if no Forced Circulation RCS flow exists.

C. To minimize heat input to the RCS.

D. The SG depressurization will lead to a loss of RCP support conditions

Surry

References:
ND-95.3-LP-38, Response to inadequate Core Cooling, Rev. 8
FR-C.1, Response to Inadequate Core Cooling, Rev. 18

Distractor Analysis:
A. Incorrect because securing RCPs is necessary because the depressurization will result in losing the RCP seal support conditions, which could damage the RCPs.
B. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS or forced flow.
C. Incorrect because the basis for securing RCPs is not associated with heat input into the RCS.
D. Correct because this is the stated reason in NB-95.3-LP-38. Losing #1 Seal support conditions could result in damage to the RCPs.

074 Inad. Core Cooling
E06EK3.1: Knowledge of the reasons for the following responses as they apply to (Degraded Core Cooling): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

Surry Requal Exam Bank Question #467
Answer: D

72. WE08G2.1.7 1

The following Unit 1 conditions exist:
- Reactor power is 58% and rising
- RCS pressure is at 2210 psig and slowly lowering
- Tavg is 557 °F and slowly lowering
- Pressurizer level is slowly lowering
- Turbine load is stable at 400 MW
- ~~SG levels are at 46% NR~~
- SG pressures are at 970 psig and slowly lowering
- Containment pressure is 9.5 psia and slowly rising
- Condenser Air Ejector RM reads 18 cpm

Justification – SG levels would be increasing from swell for a MSLB. Eliminate SG level from stem to avoid confusion.

Which ONE of the following correctly diagnoses the event?

A. Ruptured and faulted steam line break inside containment.

B. Steam line break inside containment.

C. LBCA inside containment.

D. Steam line break outside containment.

Surry

References:
General operator knowledge.

Bistractor Analysis:
A. Incorrect because although there are parameters to support the steam line break, there are no parameters to support a SGTR. Plausible because Condenser Air Ejector RM reading is given, but the value is not representative of a SGTR.
B. Correct because reactor power and ctmt pressure are rising; RCS pressure, Tavg, and SG pressures are lowering. These are ail indicative of a steam line break inside ctmt.
C. Incorrect because reactor power would not be rising during a LOCA as it would during a steam line break. Plausible because many of the parameters coincide with a LOCA.
D. Incorrect because ctmt pressure is rising. Plausible because of the aforementioned parameters that are indicative of a steam line break.

WE08 RCS Overcooling
G2.1.7: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Surry Requal Bank Question #177 (ID: EOP0076)

Answer: B
Need facility to help verify. Where is the guidance to immediately isolate a faulted / ruptured SG?

E-3 Step 3 isolates a ruptured SG. The CAUTION prior to step 4 says you should isolate feed flow in Step 4 to a faulted/ruptured SG even if level is not greater than 12%.

73. WE11EA1.2 1

The following conditions exist:
- LOCA has occurred.
- RWST level = 13% and decreasing.
- Recirculation Mode Transfer (RMT) keyswitch is in RMT Mode
- White RMT Status Light is lit.
- Amber RMT Status bight is lit.

1-SI-MOV-1860A (LHSI Suction from Sump) opens fully and 1-SI-MOV-1860B (LHSI Suction from Sump) strokes to 50% open where it trips on thermal overload. Which ONE of the following gives the correct status of Safety Injection?

A. "B" LHSI pump from the RWST is injecting into the cold legs and HHSI from LHSI pump discharge is injecting into the cold legs.

B. No Safety Injection is injecting water to the cold legs.

C. HHSI directly from the RWST (not from LHSI discharge) is injecting into the cold legs, but no LHSI is injecting into the cold legs.

D. "A" LHSI pump from the RWST and HHSI directly from the RWST (not from LHSI discharge) is being injected into the cold legs.

The interlock being tested in this question is train specific. The "A" LHSI pump would be taking suction from the containment sump and the " 5 LHSI pump would be taking suction from the RWST until it empties.
Surry

References:
ND-91.3-LP-3, Safety Injection System Operations, Rev. 15
1-E§-4.3, Transfer to Cold Leg Recirculation, Rev. 11

Bistractor Analysis:
A. Correct because 1-SI-MOV-1862A&B will not close until 1-SI-MOV-1860A&B open due to an interlock.
B. Incorrect because RWST is stili the suction source to the LHSI pumps.
C. Incorrect because LHSI Pumps are taking suction from the RWST and injecting into the cold legs and HHSI is not taking suction directly from the RWST.
D. Incorrect because HHSI is not taking suction directly from the RWST. HHSI is taking suction on the discharge of the LHSI Pumps.

WE11
EA1.2: Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation): Operating behavior characteristics of the facility.
Answer: A

74. WE12EK2.2 1

A steam break has occurred and all Steam Generators are faulted,

Which **ONE** of the following is the basis for maintaining a minimum of 60 gpm AFW flow to each Steam Generator per ECA-2.1, Uncontrolled Depressurization of All Steam Generators?

A. 60 gpm is needed to meet minimum heat sink flow requirements

B. 60 gpm to each Steam Generator will ensure even thermal hydraulic distribution across the core.

C. 60 gprn is the minimum indicated flow rate to prevent Steam Generator dryout.

D. 60 gpm is the minimum indicated flow that will ensure the feed lines stay warm to prevent excessive thermal shock to the feed lines during recovery actions.

Surry

References:
ND-95.3-LP-22, ECA-2.1 Uncontrolled Depressurization of **All** Steam Generators, Rev. 9
1-E-3, ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 16

Distractor Analysis:
A. Incorrect because this requirement is not based on minimum heat sink flow requirements, it is based on SG dryout.
B. Incorrect because this requirement is not baaed on thermal hydraulic distribution across the core. It is based on S/G dryout.
C. Correct because 60 gpm is the minimum verifiable flow rate to a steam generator. This ensures a nominal flow rate of 25 gpm to the S/G, considering detector uncertainties, to prevent dryout and thermal shock to the S/G.
D. Incorrect because the concern is with thermal shock to the SG if AFW flow rates are raised.

040 (W/E12) Steam Line Rupture - Excessive Heat Transfer
EK2.2: Knowledge of the interrelations between the (Uncontrolled Depressurization of All Steam Generators) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation *of* these systems to the operation of the facility.

Modified Surry ILT Bank Question #1010
Answer: C

75. WE13EK2.1 1

1-E-3, Steam Generator Tube Rupture, has been entered due to a ruptured tube in the "A" Steam Generator. The Team is performing Step **4,** which directs "A" Steam Generator Narrow Range SG Level to be greater than 12% prior to stopping feed flow.
Which ONE of the following correctly states the basis for this step?

A. To ensure that the ruptured steam generator tubes are covered to promote thermal stratification.

B. To ensure thermal gradients across the tubes of the ruptured steam generator do not exacerbate existing tube damage.

C. To ensure sufficient heat sink for reactor coolant system cooldown.

D. To prevent excessive primary to secondary leakage.

Surry

References:
1-E-3, Steam Generator Tube Rupture, Rev. 25
ND-95.3-LP-13, E-3 Steam Generator Tube Rupture, Rev. 11

Distractor Analysis:
A. Correct because this **is** the basis as stated in MD-95.3-LP-13.
B. incorrect because the concern is not thermal gradients across the tubes. The concern is to cover the tubes for thermal stratification and then stop AFW flow as soon as the tubes are covered to give margin to overfill, while mitigating release to the public.
C. Incorrect because this SG will not be used for the RCS cooldown.
D. Incorrect because the dP is still going to induce leakage even at 12% SG level.

WE14 Steam Generator Over-pressure
EK2.1: Knowledge of the interrelations between the (Steam Generator Overpressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question is modified from a Braidwood Question.
Answer: **A**

76. 00162.4.30 1

Which ONE *of* the following states an event that *is* required to be reported *to* the NRC within **1** hour of discovery?

A. An inadvertant Safety Injection due to an instrument surveillance error.

B. The Shift Supervisor authorizes the individual insertion of control rods into the core without bank overlap to shutdown the reactor in an emergency.

C. **A** hypochlorite spill outside the Polishing Building of which the EPA has been notified.

D. A radioactive release such that if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

Surry

References:
VPAP-2802, Notifications and Reports, Rev. 17.

Distractor Analysis:
A. Incorrect because this is a **4** hour reportable event. Plausible because the applicant may think that inadvertant safety injection is important enough to require reporting to the NRC within one hour.
B. Correct per VPAP-2802 Section 6.3.3 for deviation from Tech Specs. (VPAP-2802 Page 77.)
C. Incorrect because this is a **4** hour reportable event. Plausible because the applicant may think that a hypochlorite spill with EQA notification **is** important enough to require reporting to the NRC within one hour.
D. Incorrect because this is a 24 hour reportable event. Plausible because the applicant may think that a large radioactive release is important enough to require reporting to the NRC within one hour.

001 Control Rod Drive
G2.4.30 Knowledge of which events related to system operations / status should be reported to outside agencies.
Answer: B

Memorization of this information is not required knowledge. The SROs must be given a copy of VPAP-2802 as a reference.

77. 00462.1.32 **2**

During Unit 1 REFUELING SHUTDOWN and COLD SHUTDOWN operations, the following valves shall be locked, sealed, or otherwise secured in the closed position except during planned dilution or makeup activities.
- 1-CH-223, or
- 1-CH-212, 1-CH-215, and 1-CK-218

Which ONE of the following correctly describes the time requirement and reason for locking, sealing, or otherwise securing these valves following a planned dilution or makeup activity in accordance with Technical Specifications?

A. 15 minutes to prevent inadvertant boron dilution of the WCS.

B. 60 minutes to ensure the proper safety system alignment.

C. 15 minutes to ensure the proper safety system alignment.

D. 60 minutes to prevent inadvertant boron dilution of the RCS

Surly

References:
Technical Specification 3.2.E.3, Amendment 199

Distractor Analysis:
A. Correct per Technical Specifications end Basis.
B. Incorrect because Technical Specifications require within 15 minutes.
C. Incorrect because Technical Specifications Basis states that these valves shall be closed to provide assurance that an inadvertant boron dilution will not occur.
D. Incorrect because Technical Specifications require within 15 minutes.

004 Chemical and Volume Control
G2.1.32: Ability to explain and apply all system limits and precautions.

Answer: A

78. 609EA2.39 1

Given the following Unit **1** conditions:
- **A** small break LOCA has occurred
- As directed by the EOPs, the RCPs have been tripped
- 1-ES-1.2, Post-LOCA Cooldown and Depressurization, Step 20, "Verify Natural Circulation," is being performed
- RCS pressure is 1496 psig
- Wide Range T-Cold indications are 505 ^{o}F and slowly decreasing
- Wide Range T-Hot indications are 515 ^{o}F and slowly decreasing
- CETCs are 581 ^{o}F end stable
- containment Pressure is 10 psia
- Containment Radiation bevels are: 5.0 x 10^{5} R/hr
- SG Narrow Range Levels are: A=22%, B=24%, C=22%, and slowly decreasing
- SG Pressures are 715 psig and stable
- RVL!S Full Range = 50%

According to 1-E§-1.2, which ONE of the following correctly stales the status of Natural Circulation end the correct operator actions?

A. Natural Circulation criteria are met. Begin depressurizing when subcooling is > 85 ^{o}F.

B. Natural Circulation criteria are not met due to CETCs not decreasing. Depressurize the SGs by raising steam flow rate through the steam dumps. Then depressurize when subcooling is > 95 ^{o}F.

C. Natural Circulation criteria are not met due to SG pressure parameters not satisfied. Depressurize the SGs by raising steam flow rate through the steam dumps. Then depressurize when subcooling is > 85 ^{o}F.

D. Natural Circulation criteria are not met due to inadequate subcooling. Cool the RCS by raising steam flow rate through the steam dumps. Then depressurize when subcooling is > 95 ^{o}F.

Surry

References:
1-ES-1.2, Post LQCA Cooldown and Depressurization, Rev. 21

Distractor Analysis:
A. Incorrect because there is not adequate subcooling.
B. Incorrect because CETCs do not need to be decreasing
C. Incorrect because SG parameters are satisfied.
D. Correct because there is inadequate subcooling (16 ^{o}F < 85 ^{o}F). ES-1.2 Step 20 RNQ directs dumping of more steam. The basis for Step 21 of dumping steam until subcooling is < 95 ^{o}F is to ensure that the 85 ^{o}F natural circ criteria is not violated. The Degraded Containment numbers were used due to the CETC = 581^{o}F; P = 1490 psig = 1505 psia; Tsat(1505 psia) = 597 ^{o}F; Subcooling = 597 - 581 = 16^{o}F

Surry ILT Bank Exam Question #1069

009 Small Break LOCA
EA2.39: Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling.
Answer: D

79. 025G2.4.7 1

The following Unit 1 conditions exist:
- RCS is not pressurized
- RCS level is 16.00 feet as read on 1-RC-LI-100A

Which ONE of the following specifies the mimimum mandatory backup cooling method(s) required to be available before entering the above plant conditions, in accordance with OSP-ZZ-004, Unit 1 Safety Systems Status List For Cold Shutdown / Refueling Conditions?

A. Reflux Boiling **AND** Gravity Feed and Bleed.

B. Gravity Feed and Bleed ONLY.

C. Forced Feed and Bleed AND Gravity Feed and Bleed.

D. Forced Feed and Bleed ONLY.

Surry

References:
1-OSP-ZZ-004, Unit 1 Safety Systems Status List For Cold Shutdown / Refueling Conditions, Rev. 27
1-AP-27.00, **Loss** of Decay Heat Removal Capability, Rev. 10
ND-95.2-LP-12, Loss of RHR Events, Rev. 9

Distractor Analysis:
A. Incorrect: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.
B. Incorrect: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.
C. Incorrect: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.
D. Correct: Per 1-OSP-ZZ-004, Step 6.1.2, Forced Feed and Bleed is the only Mandatory Backup method required.

025 Loss of RHR
G2.4.7: Knowledge of event based EOP mitigation strategy

Answer: D

Memorization of Backup Cooling methods for each of the plant conditions in 1-OSP-ZZ-004 is not required knowledge. SROs must **be** given 1-OSP-ZZ-004 as a reference

80. 055EA2.03 1

The following conditions exist:
- A loss of all AC power has occurred.
- The STA reports the status of the CSFs are as follows:
  - Subcriticality - RED
  - Core Cooling - RED
  - Heat Sink - WED
  - Integrity - GREEN
  - Containment - GREEN
  - Inventory - YELLOW

Which ONE of the following proceures should be used to mitigate these conditions?

A. 1-FR-S.1, Response to Nuclear Power Generation / ATWS

B. 1-ECA-0.0, Loss of All AC Power

C. 1-FR-H.1, Response to Lass of Secondary Heat Sink

D. 1-FR-C.1, Response to Inadequate Core Cooling

Surry

References:
1-ECA-0.0, Loss of All AC Power, Rev. 24

Distractor Analysis:
A. Incorrect because FR's should not be implemented while in ECA-0.0. (see NOTE prior to step 1 of ECA-0.0)
B. Correct because this is the correct procedure to mitigate the loss of ac power.
C. Incorrect because FR's should not be implemented while in ECA-0.0.
D. Incorrect because FR's should not be implemented while in ECA-0.0.

Surry ILT Exam Bank Question 15899

055 Station Blackout
EA2.03: Ability to determine or interpret the following as they apply to Station Blackout: Actions necessary to restore power.
Answer: B

81. 05662.4.45 1

The following Unit 1 conditions **exist:**
- Power = 108%
- Condenser vacuum is lowering slowly.
- Steam Generator levels are 45% and lowering.
- Several alarms have annunciated, including:
  - 1H-G8, FW PP DISCH HDR LO PRESS
  - 1J-G4, CN PPS DISCH HDR LO PRESS
  - 1C-A1, RCP 1A CC RETURN LO FLOW
  - 1C-B1, RCP 1 B 66 RETURN LO FLOW
  - 1C-C1. RCP I C CC RETURN **LO** FLOW

Which ONE of the following states the SRO's correct prioritization of the above conditions as indicated by the procedures and actions chosen to mitigate or correct the conditions?

A. Trip the Reactor followed by tripping the Reactor Coolant Pumps. Enter E-0, Reactor Trip or Safety Injection.

B. Enter AP-10.05, boss of Semi-vital Bus. Verify that the standby condensate pump has started and reduce turbine load.

C. Enter AP-21.00, boss of Main Feedwater Flow. Maintain full power operation and manually control Steam Generator levels by placing Feedwater Regulating Valves in MANUAL control.

D. Enter AP-23.00, Rapid Load Reduction, to bring the unit offline, followed by tripping the Reactor Coolant Pumps.

Surry

References:
NB-90.3-LP-5, Vital end Semi-vital Bus Distribution, Rev. 11
1-AP-10.05, Loss of Semi-vital **Bus,** Rev. 16
1-A$-21.00, boss of Main Feedwater Flow, Rev. 5
1-AP-23.00, Rapid Load Reduction, Rev. 15
1H-G8, FW PP DISCH HDR LO PRESS, Rev. 0
1J-G4, CN PPS DISCH HDR LO PRESS, Rev. 0
1C-A1, RCP 1A CC RETURN LO FLOW, Rev. 2
1C-B1, RCP 1B CC RETURN LO FLOW, Rev. 2
1C-C1, RCP I C CC RETURN LO FLOW, Rev. 2

Distractor Analysis:
A. Incorrect because loss of SVB causes indication to be lust for RCP CC Flow indication. RCPs should not be tripped. Plausible because if RCPs actually had no cooling, the **Rx** should be tripped and RCPs should be secured.
B. Correct because all indications in the stem are caused by a loss of SVB. Verifying S/B condensate Pump **starts** and turbine load reduction are correct per AP-10.05.
C. Incorrect because maintaining load at 100% will cause SG levels to continue to go down. The FW and Condensate Recircs have failed open on the **loss** of the SVB, thus making a load reduction a necesity. Plausible because SG levels are lowering and an Applicant may think that opening a FRV may help to mitigate the condition.
D. Incorrect because the unit should not be taken off line using AP-23.00 and RCPs should not be tripped due to the loss of the SVB. Plausible because rapidly bringing the unit off line and securing RCPs, given the stated conditions, may appear logical to the applicant.

Modified Sur9 ILT Exam Bank Question #224 (maybe it could be considered a new question?)

056 Condensate
G2.4.45: Ability to prioritize and interpret the significance of each annunciator or alarm.
Answer: B

82. 057G2.1.6 1

The following Unit I conditions exist:
- Reactor Power = 30%
- Plant is in a Chemistry hold during a power ascension
- A loss of Vital Bus III occurs and operators enter 1-AP-10.03, Loss of Vital Bus III
- Electricians quickly find a fault on Vital Bus 1-III and believe that it will take 10 hours *to* repair.
- 1-CC-TV-105A, CCW TV for the "A" Reactor Coolant Pump (RCP), has closed and cannot be re-opened.
- RCP temperatures are starting to slowly rise.

Which ONE of the following set of actions should the Senior Reactor Operator (SRO) direct given the above conditions?

A. The SRO should direct the securing of the "A" RCP. Reactor power may be maintained at 30% for the duration of the 10 hour repair *to* re-energize Vital Bus 1-III.

B. The SRO should direct the securing of the " A RCP. Reactor power may be maintained at 30% for two hours, at which time the SRO should direct preparation to bring the unit *to* hot shutdown within the following six hours.

C. The SRO should direct a Reactor Trip, followed **by** the securing of the "A" RCP. The SRO should then direct performance of 2-E-0, Reactor Trip or Safety Injection, and continue with applicable actions of 1-AP-10.03.

D. The SRO should direct a controlled plant shutdown. If RCP temperatures exceed action level limits, the pump should ***be*** secured and the SRO should direct continuation of the controlled plant shutdown.

Surry

References:
ND-93.3-LP-16, Permissive/Bypass?Trip Status Lights. Rev. 8
ND-93.3-LP-10, Reactor Protection - General, Rev. *5*
ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11
1-AP-10.03, Loss of Vital Bus III, Rev. 8

Bistractor Analysis:
A. Incorrect because TS 3.16 and commitments made in GL-91-11 (also located in Note prior to Step 17 in AP-10.03). The VB must be re-powered within 2 hours, or the unit must be in HSD within the next 6 hours. Also incorrect because AP-10.03 will require a reactor trip. Plausible because the loss of VB causes a loss of cooling to "A" RCP. It may appear OK *to* continue operation because the power is < P-8.
B. Incorrect because AP-10.03 requires a reactor trip and securing of RCP if CCW will not be restored prior to RCP temperatures reaching action level limits. Plausible because of the NOTE mentioned in the previous distractor analysis.
C. Correct because AP-10.03 directs Rx Trip and securing of RCP if CCW will not be restored prior to getting cooling back to that pump. The stern states that the TV is closed and cannot be re-opened, thus preventing cooling *to* be restored *to* the RCP.
D. Incorrect because AP-10.03 directs Rx Trip, not a controlled shutdown. Plausible because power is < P-8, which may allow the applicant to incorrectly believe that a shutdown is acceptable.

057 Loss of Vital AC Inst Bus
G2.1.6: **Ability** to supervise and assume a management role during plant transients and upset conditions.
Answer: C

83. 05862.4.32 4

The fullowing Unit 1 conditions exist:
- Unit 1 power is 100%
- No annunciators are lit
- Annunciator 1K-H1, Hathaway Power Available Lights, has just extinguished |

Which ONE of the following is the correct Abnormal Procedure to enter and correct Event Classification? (Reference provided)

A. Enter 0-AP-10.43, boss of Main Control Room Annunciators, due to the loss of one of the power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare a Notification of Unusual Event if the loss of annunciators lasts for greater than 15 minutes.

B. Enter 0-AP-10.13, Loss of Main Control Room Annunciators, due to the loss of both power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare a Notification of Unusual Event if the loss of annunciators lasts fur greater than 15 minutes.

C. Enter 1-AP-10.06, Loss of DC Power, and 0-AP-10.13, Loss of Main Control Room Annunciators, due to a loss of DC power and loss of one of the power supplies to Unit 1 annunciators. Enter the Emergency Plan and declare an Alert if the loss of annunciators lasts for greater than 15 minutes.

D. Enter 1-AP-10.06, Loss of DC Power, and 0-AP-10.13, boss of Main Control Room Annunciators, due *to* ***a*** loss of DC power and loss of both power supplies to Unit **1** annunciators. Enter the Emergency Plan and declare an Alert if the loss of annunciators lasts for greater than 15 minutes.

Surry

References:
0-AP-10.13, Loss of Main Control Board Room Annunciators, Rev. **4**
EPIP-1.01, Emergency Manager Controlling Procedure, Rev. 43

Distractor Analysis:
A. Incorrect because 1K-H1 not lit is indication of both power supplies to Unit 1 annunciator Panels having been lost.
B. Correct because 1K-H1 not lit is indication of both power supplies to Unit 1 annunciator Panels having been lost. EPIP-1.01 Page 6 states that if safety system annunciators are lost for greater than 15 minutes while above CSD, then a NOUE shall be declared.
C. Incorrect because 1K-H1 not lit is indication of both power supplies to Unit 1 annunciator Panels having been lost. Since the plant is still at 100% power, there is no indication that any DC Bus has been lost; therefore 1-AP-10.06 should not be entered. An Alert classification based on the loss of **DC** would be incorrect. **As** stated above, a NOUE is the correct classification.
D. Incorrect because the plant is still at 100% power, there is no indication that any DC Bus has been lost; therefore 1-AP-10.06 should not be entered. An Alert classification based on the loss of DC would be incorrect. As stated above, a NOUE is the correct classification.

Provide EPIP-1.01 Pages 6 and 27

058 Loss of DC Power
G2.4.32: Knowledge of operator response to a loss of all annunciators.

Answer: B

84. 062A2.12 1

Unit 1 is at 100% power. It experiences a loss of Vital Bus I at 1200 hours on Monday. Operators enter 1-AP-10.01, Loss of Vital Bus I, and re-energize the Vital Bus from its alternate source at 1215 hours on Monday.

Which ONE of the following correctly states the required actions based on the above condition?

A. In accordance with 1-AP-10.01, Vital Bus I must be re-energized from its primary source by 1400 hours on Monday, or be in Hot Shutdown by 2000 hours on Monday.

B. In accordance with 1-AP-10.01, Vital Bus I must be re-energized from its primary source by 1415 hours on Monday, or be in Hot Shutdown by 2015 hours on Monday.

C. In accordance with 1-AP-10.01, Vital Bus I must be re-energized from its primary source by 1200 hours on Tuesday, or be in Hot Shutdown by 1800 hours on Tuesday.

D. No shutdown requirements are in effect as long as Vital Bus I is energized.

Surry

References:
1-AP-10.01, Loss of Vital Bus I, Rev. 13
ND-90.3-LP-5, Vital and Semi-vital Bus Distribution, Rev. 11

Distractor Analysis:
A. Incorrect because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because if the bus is not energized, it must be repowered within 2 hours and 1400 hours is 2 hours after 1200 hours.
B. Incorrect because per AP-70.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because if the bus is not energized, it must be repowered within 2 hours and 1415 hours is 2 hours after 1215 hours.
C. Correct because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSB within the next 6 hours (also see ND-90.3-LP-5 Page 15). The consequences of having VB-I not energized by its primary source are mitigated, or corrected, by ensuring that it is energized from its primary source within the specified time requirement.
D. Incorrect because per AP-10.01 Step 16 c, the VB must be powered from its normal source within 24 hours or the unit must be placed in HSD within the next 6 hours (also see ND-90.3-LP-5 Page 15). Plausible because the Vital Bus is energized and the plant would be operating satisfactorily.

062 AC Electrical Distribution
A2.12: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Restoration of power to a system with a fault on it.

Answer: C

85. 062AA2.04 1

The following Unit 1 conditions exist:
- Power = 100%
- 1-CH-P-1A Charging Pump is operating
- 1-SW-P-10A Charging Pump Service Water Pump is operating
- 1-SW-P-10B Charging Pump Service Water Pump is in standby
- 1D-G5, SW OR CC PPS DISCH TO CHG PPS LO PRESS, alarms
- 1-CH-P-1A Charging Pump Bearing Temperature = 175 °F
- I-CH-P-1A Charging Pump Oil Cooler Outlet Temperature = 150 °F
- The Pressure Indication on the discharge of 1-SW-P-10A Charging Pump Service Water Pump (SW-PI-26) reached a minimum value of 10 psig where it remains stable.
- The Operatw in the field reports back to the Control Room that 1-SW-P-10A Charging Pump Service Water Pump is noisy and has high vibrations.

Which ONE of the following correctly states the appropriate assessment of the above conditions and appropriate operator action based on that assessment?

A. Bearing Temperature is not within limits. The " A Charging Pump is INOPERABLE. Direct starting standby Charging Pump Service Water Pump, direct securing the " A Charging Pump Service Water Pump, and notify the System Engineer.

B. Bearing Temperature is not within limits. 1-CH-P-1A Charging Pump is INOPERABLE. Verify auto start of 1-SW-P-10B Charging Pump Service Water Pump, and notify the System Engineer.

C. Oil Cooler Outlet Temperature is not within normal operating band. 4-CH-P-ZA Charging Pump is OPERABLE. Direct starting standby Charging Pump Service Water Pump, direct securing the "A" Charging Pump Service Water Pump, and notify the System Engineer.

D. Oil Cooler Outlet Temperature is not within normal operating band. Performance of Charging Pump Operability and Performance Test for 1-CH-P-1A Charging Pump must be directed to determine OPERABILITY.

Surry

References:
1D-G5, SW OR CC PPS DISCH TO CHG PPS LO PRESS, Rev. 3
11448-FM-071B, Sh. 1 of 2, Flow / Valve Operating Numbers Diagram, Circulating and Service Water System, Surry Power Station Unit 1, Virginia Power, Rev. 50.
NB-89.5-LP-2, Service Water System, Rev. 20
1-OP-CH-002, Charging Pump A Operations, Rev. 43
1-OPT-CH-001, Charging Pump Operability and Performance Test For 1-CH-P-1A, Rev. 33

Distractor Analysis:
A. Incorrect because Bearing Temperature is less than 180 °F. OPT-CH-001 Pg 9 states that the upper admin limit is 180 °F. The Charging Pump is still OPERABLE.
B. Incorrect because Bearing Temperature is less than 180 °F. OPT-CH-004 Pg 9 states that the upper admin limit is 180 °F. The standby pump will not start until 8 psig.
C. Correct because Oil Cooler Outlet Temperature is not within the normal operating band (80 - 120 °F) as states in OPT-CH-001. However, the problem is not with the Charging Pump, **but** with the Service Water flow. so swapping Charging Pump Service Water Pumps is the correct initial

action based on the AWP.
D. Incorrect because there ***is*** no indication that the Charging Pump has a problem. Given the above alarm, all indications suggest that the problem is with the Service Water flow. Therefore, performance of the Operability and Performance Test **for** the Charging Pump would serve no purpose.

062 Loss of svc Water
AA2.04: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The normal values and upper limits for the temperatures of the components cooled by SWS.
Answer: C

86. 07902.4.48 1

The fullowing Unit 1 conditions exist:
- Reactor Power = 100%
- A loss of Containment Instrument Air has occurred
- 1B-F6, CTMT INST AIR HDK LO PRESSURE, annunciates
- 1D-C6, PRZR PWR RELIEF VV LO AIR PRESS, annunciates
- Containment Instrument Air Pressure = 75 psig
- Containment Instrument Air was crosstied with Turbine Building Instrument Air and is still reading 75 psig.

Which ONE of the following operator actions is required?

A. Both Pressurizer PORVs are operable following the crosstie. Verify the operability by closing PORV Block Valves, stroking PORVs, then re-opening the PORV Block Valves.

5. Both Pressurizer PORVs are operable following the crosstie. Ne further action associated with the PORVs is required.

C. Declare both Pressurizer PORVs inoperable. Close and remove power from both PORV block valves within one hour and be in HSD within 6 hours.

D. Declare both Pressurizer PORVs inoperable. Close, but leave energized, both PORV block valves within one hour and be in HSB within 6 hours.

Surry

References:
ND-92.1-LP-1, Station Air Systems, Rev. 43
ND-88.1-LP-3, Pressurizer and Pressure Relief, Rev. 12
1B-F6, CTMT INST AIR HDR LO PRESS, Rev. 1
1D-C6, PRZR PWR RELIEF VV LO AIR PRESS. Rev. 4
Technical Specification 3.1.A.6.c, Reactor Coolant System / Relief Valves

Distractor Analysis:
A. Incorrect because (per 1D-C6) with CTMT Inst Air P < 80 psig, the PORVs are inoperable.
B. Incorrect because (per 1D-C6) with CTMT Inst Air P < 80 psig, the PORVs are inoperable.
C. Correct because PORVs are not capable of being manually cycled with CTMT Inst Air P < 80 psig. Therefore, within 1 hour. the CTMT Inst Air pressure must be > 80 psig or the block valves must be closed and de-energized. Furthermore, the plant must be in HSD within the next 6 hours.
D. Incorrect because, as stated in "C" above, power must be removed from the block valves.

Surry Requal Bank Question #394 (ARP0001)

079 Station Air
G2.4.48: Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.
Answer: C

SROs must be given a copy of ARP D-C-6. This question was taken from the REQUAL Exam Bank which is an open reference exam.

87. 103A2.01 1

The following Unit 1 conditions exist:
- Plant is in Mode 1
- Personnel Airlock Seal Leakage Testing has just been completed
- The Personnel Airlock Inner Boor Seal exceeded Technical Specifications leakage limits
- Earlier in the year the Personnel Airlock Inner Boor exceeded Technical Specifications leakage limits and the Personnel Airlock Outer Door was opened for a total of **50** minutes during the inoperability of the Personnel Airlock Inner Door

Which **ONE** of the following actions would satisfy required Technical Specification Actions **for** the Personnel Airlock Doors?

A. The Personnel Airlock Outer Door may not be opened to pursue the repair and retest. The plant must be shutdown and cooled down per Plant General Operating Procedures. The plant must be in Hot Shutdown within 6 hours and Cold Shutdown within the following 30 hours.

B. The Personnel Airlock Outer Doer may be opened for 10 minutes to pursue the repair and retest of the Personnel Airlock Inner Boor Seal. Per VPAP-0106, Subatmospheric Containment Entry, the Shift Supervisor shall supervise the containment entry and exit process.

C. The Personnel Airlock Outer Door may be opened for **15** minutes to pursue the repair and retest of the Personnel Airlock Inner Door Seal. Per VPAP-0106, Subatmospheric Containment Entry, the Unit SRO shall supervise the containment entry and exit process.

D. The Personnel Airlock Outer Boor may be opened for 1 hour to pursue the repair and retest of the Personnel Airlock Inner Door Seal. Per $/PAP-0106, Subatmospheric Containment Entry, the Unit SRO shall supervise the containment entry and exit precess.

Surry

References:
VPAP-0106, Subatmospheric Containment Entry, Rev. **5**
Technical Specifications **3.8,** Containment (Amendments 172 and **171);** 1.0.G, Definitions (Amendment **180)**

Distractor Analysis:
A. Incorrect because the Outer Door may be opened for 10 minutes since it has already been opened 50 minutes this year while the inner door was inoperable.
B. Correct because per Tech Specs, the Outer Door may **be** opened for 15 minutes or 60 minutes for the year (which leaves 10 more minutes for this instance). Furthermore, the SS must supervise the containment entry and exit process per VPAP-0106 Section **5.1.**
C. Incorrect because the Outer Door may be opened for 10 minutes and the SS must supervise the containment entry and exit per VPAP-0106 Section **5.1.**
D. Incorrect because the Outer Door may be opened for 10 minutes.

103 Containment
A2.01: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Integrated Leak Rate Tests.
Answer: B

88. G2.1.34 1

The following Unit 1 conditions exist:
- The Unit has been operating at 100% power for the past two weeks
- Chemistry has just provided the following results from a Reactor Coolant System sample that was taken 1 hour ago:
  - RCS Chloride = 0.15 ppm
  - RCS Fluoride = 0.15 ppm
  - RCS Oxygen = 0.15 ppm

Which ONE of the following describes the above conditions and appropriate operator action?

A. Oxygen concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the plant to cold shutdown conditions.

B. Oxygen concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the oxygen concentration within limits. If the oxygen concentration is outside of the limit after 24 hours, then the plant must be taken to cold shutdown.

C. Chloride concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the plant to cold shutdown conditions.

D. Chloride concentration is above the allowable Technical Specification limit. Per Technical Specifications, corrective action must be taken immediately to bring the chloride concentration within limits. If the chloride concentration is outside of the limit after 24 hours, then the plant must be taken to cold shutdown.

Surry

References:
Technical Specifications 3.1.F.1 and Basis

Distractor Analysis:
A. Incorrect because, according to the Tech Spec Basis, the plant has **24** hours to see if their corrective actions will bring the parameter within spec. If after 24 hours the parameter is not within spec, then the plant must be taken to cold shutdown using normal plant procedures.
B. Correct because, according to the Tech Spec Basis, the plant has 24 hours to see if their corrective actions will bring the parameter within spec. If after 24 hours the parameter is not within spec, then the plant must be taken to cold shutdown using normal plant procedures.
C. Incorrect because Chloride Concentration is within limits.
D. Incorrect because Chloride concentration is within limits.

G2.1.34: Ability to maintain primary and secondary plant chemistry within allowable limits.
Answer: B

89. G2.1.4 1

The following plant conditions exist:
- Unit 1 is shutdown and subcritical by 5.35% delta k / k
- Unit 1 Tavg is 100 °F
- Unit 2 is shutdown and subcritical by 2.35% delta k / k
- Unit 2 Tavg is 190 °F

Which ONE of the following correctly states the MINIMUM shifl crew composition per Technical Specifications?

A. 1 SS, 1 Unit SRO, 3 ROs, 4 AOs, and 1 STA

B. 1 SS, 2 Unit SROs, 3 ROs, 4 AOs, and no STA.

C. 1 SS, 1 Unit SRO, 3 ROs, 4 AOs, and no STA.

D. 1 SS, no Unit SRO, 2 ROs, 4 AOs, and no STA.

Surry

References:
Technical Specification Table 6.1-1 (Minimum Shift Crew Composition), Amendment No. 123.

Distractor Analysis:
A. Incurrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
B. Incorrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
C. Incorrect because it does not match the minimum requirements for one unit in Cold Shutdown and one unit in Refueling Shutdown.
D. Correct because it matches the requirement for one unit in Cold Shutdown and one unit in Refueling Shutdown.

G2.1.4
Knowledge of shifl staffing requirements
Answer: D

90. G2.2.23 1

The foliowing conditions exist:

- Unit 1 is at 50% power
- Unit 2 is in startup mode with Tavg = 410°F
- Unit *2* Steam Briven AFW Pump and one Motor Driven AFW Pump are declared *to* be inoperable at 0800 hours on August 11 (all other AFW equipment is operable)
- Unit 2 Motor Driven AFW Pump is restored to operable status at 1100 hours and Unit 2 Tavg = **410°F**

Which ONE of the following sets of Technical Specification actions is correct? (Reference provided)

A. Initially (with both pumps inoperable) both AFW Pumps must be restored or Unit 2 must not enter Hot Shutdown. All Unit 1 Technical Specification Actions will be less restrictive than the Unit 2 Technical Specification Actions.

B. Unit 2 AFW actions do not apply. Initially (with both pumps inoperable) Unit 1 must be in Hot Shutdown by 08/25 at 1400 hours and Cold Shutdown by 08/26 at 2000 hours. After the Motor Driven AFW Pump is operable no Unit 1 actions would be in effect.

C. Initially (with both pumps inoperable) Unit 2 must be in Cold Shutdown by 08/12 at ~~2000~~ 1400 hours and restore either AFW pump by 08/25 at 0800 hours or Unit 1 must be placed in Hot Shutdown by 08/25 at 1400 hours. After the Motor Driven AFW Pump ~~us restored, the Steam Driven AFW Pump must be restored by 08/14 at 0800 hours or Unit 2 must be in Hot Shutdown by 08/14 at 2000 hours~~ Is restored, all AFW Tech Spec clocks are exited.

D. initially (with both pumps inoperable) Unit *2* shall not enter Hot Shutdown and must be in Cold Shutdown by 08/12 *at* 2000 hours and restore either AFW pump within 14 days or Unit 1 must be placed in Hot Shutdown by 08/25 at 0800 hours. After the Motor Driven AFW Pump is restored, the Steam Driven **AFW** Pump must also be restored by 08/14 at 0800 hours or Unit 1 must be in Hot Shutdown by 08/14 at 2000 hours.

Justification – The Steam Driven AFW pump is not required to be operable until 10% power. The clock for Unit 2 is not the standard 6hours to HSD and 30 hours to CSD since the Unit is already below HSD. The Unit is in a 30 hour clock to CSD.

Surry

References:
Technical Specifications 3.6.C, 3.6.F, 3.6.G, and 3.01

Distractor Analysis:
A. Incorrect because Unit 2 does not need to be placed in HSD until 12 hours following 08114 at 0800 hours.
B. Incorrect because Unit 2 Technical Specification Actions do apply above 350 °F and 450 psig.
C. Correct because LCO 3.0.1 is entered with both pumps inoperable because there is not a tech spec condition that covers this situation. Once the MDAFW Pump is operable, LCO 3.0.1 is exited, but 3.6.F and 3.6.C still applies for Unit 2.
D. Incorrect because Unit 1 does not need to be shutdown with only the Unit 2 Steam Driven AFW Pump inoperable.

G2.2.23
Ability to track limiting conditions for operations.

Answer: C

91. G2.2.31 1

Unit 1 has been shut down for 21 days and fuel movement has just commenced. Which ONE of the following is correct with regard to Fuel Building Exhaust and Containment Purge Exhaust?

A. Fuel Building Exhaust and Containment Purge Exhaust must be manually aligned to continuously pass through CAT1 filters during fuel movements.

B. Fuel Building Exhaust and Containment Purge Exhaust will automatically align to the CAT1 filters if a fuel handling accident occurred at this time.

C. There is no need to manually align Fuel Building Exhaust or Containment Purge Exhaust to the CAT1 filters because the fuel has decayed for a sufficient period of time such that radiological consequences from a fuel handling accident would be acceptable without iodine filtration.

D. Fuel Building Exhaust and Containment Purge Exhaust must be secured during fuel movements to prevent automatically tripping the purge.

Surry

References:
ND-92.5-LP-7, Refueling Abnormal Procedures, Rev. 10

Distractor Analysis:
A. Correct because the automatic alignment feature is bypassed when fuel has decayed **for less** than 30 days. Therefore, it must be manually aligned prior to moving fuel.
B. Incorrect because the automatic alignment feature is bypassed when fuel has decayed fur less than 30 days.
C. Incorrect because 30 days is considered sufficient decay time, net 21 days.
B. Incorrect because this in only a requirement during movement of the upper internals.

G2.2.31
Knowledge of procedures end limitations involved in initial core leading.
Answer: **A**

This question was developed from an outdated lesson plan. Recommend changing as follows:

Unit 1 has been shutdown for 21 days and fuel movement has just commenced. Which one of the following is correct with regard to Containment Purge Exhaust?

a. Containment Purge Exhaust shall be manually aligned to continuously pass through CAT 1 filters during fuel movement.

b. Containment Purge Exhaust shall be manually aligned to continuously pass through CAT 2 filters during fuel moveme[illegible]

c. Containment Purge Exhaust may be manually aligned to continuously pass through CAT 1 filters during fuel movement. If a containment radiation monitor high alarm isolates purge flow, fuel movement may continue after purge flow is verified isolated.

d. Conta nment Purge Exhaust may be manually aligned tc continuously pass through CAT 2 filters during fuel movement. If a containment radiation monitor high alarm isolates purge flow, fuel movement may continue after purge flow is verified isolated.

ANSWER (c)

Distractor Analysis

a. Incorrect – This was the original design for the ventilation system during fuel movement but was changed when the source term reduction study was completed.

b. Incorrect – Containment Purge is now aligned using VS-F 59 which normally flows through the CAT 2 filter. For fuel movement, a jumper is installed to flow through the CAT 1 filter.

c. Correct answer

d. Incorrect – Containment Purge is now aligned using VS-F 59 which normally flows through the CAT 2 filter. For fuel movement, a jumper is installed to flow through the CAT 1 filter.

92. G2.2.6 1

Which ONE of the following correctly states items that require a Regulatory Screen to be performed in accordance with VPAP-3001, Station and Regulatory Reviews?

A. Emergency Action Level Change AND Station Curve Changes

B. Seismic Anaiyses AND Heating-Ventilation and Air Conditioning Analyses

C. Fire Protection Plan Changes AND Plant Flood Analyses

D. Offsite Dose Calculation Manual Changes AND Equipment Qualification Analyses

Surry

References:
VPAP-3001, Station and Regulatory Reviews, Rev. 9

Distractor Analysis:
A. Incorrect because Emergency Action bevel Changes are to be processed IAW VPAP-0502 (see VPAP-3001 Page **2** of Att. 3), a Regulatory Screen is not required. Plausible because both items are listed on VPAP-3001 Att. 3 Page 2.
B. Correct per VPAP-3001 Page **2** of Att.3.
C. Incorrect because Fire Protection Plan Changes are to be performed IAW VPAP-2401 (see VPAP-3004 Page 2 of Att. 3), a Regulatory Screen is not required. Plausible because **both** items are listed on VPAP-3001 **Att.** 3 Page **2.**
D. Incorrect because ODCM changes are to be performed IAW VPAP-2103N, a Regulatory Screen is not required. Plausible because both items are listed on VPAP-3001 **Att.** 3 Page 2.

G2.2.6: Knowledge of the process for making changes in procedures as described in the safety analysis report.
Answer: B

The SRO has no control over any of these. This is not required knowledge. VPAP-3001 has no SRO responsibilities listed in it...SROs must be given a copy of VPAP-3001 for reference.

93. G2.4.29 1

Which ONE of the following are all responsibilities that shall <u>NOT</u> be delegated by the Station Emergency Manager?

A. Ordering Site Evacuation, Authorizing Emergency Exposure Limits.

B. Authorizing Notifications of NRC, State and Local Agencies of the Emergency Status, Authorizing Emergency Exposure Limits.

C. Authorizing Notifications of NRC, State and Local Agencies of the Emergency Status, Restricting Access to the Site.

D. Authorizing Emergency Exposure Limits, Restricting Access to the Site

Surry

References:
ND-95.5-LP-2, Station Emergency Manager, Rev. 8
Site Emergency Plan, Rev. 46

Distractor Analysis:
A. Incorrect because ordering a site evacuation may be delegated.
B. Correct because the answer is clearly stated in both of the references.
C. Incorrect because restricting access to the site may be delegated.
D. Incorrect because restricting access to the site rnay be delegated.

G2.4.29: Knowledge of the emergency plan.
Answer: B

94. G2.4.38 I

Which ON€ of the following correctly states the preferred order for assuming the Station Emergency Manager responsibilities from the Shift Supervisor once the Technical Support Center is activated?

A. Manger Nuclear Operations, Director Nuclear Station Safety and Licensing, Director Nuclear Station operations and Maintenance, Another Qualified SRO

B. Site Vice-president, Director Nuclear Station Safety and Licensing, Director Nuclear Station Operations and Maintenance, Manger Nuclear Operations

C. Site Vice-President, Director Nuclear Station Operations and Maintenance, Director Nuclear Station Safety and Licensing, Manger Nuclear Operations

D. Site Vice-president, Director Nuclear Station Operations and Maintenance, Manger Nuclear Operations, Director Nuclear Station Safety and Licensing

Surry

References:
ND-95.5-LP-2, Station Emergency Manager, Rev. 8

Distractor Analysis:
A. incorrect because this is not the preferred order as specified in ND-95.5-LP-2 Pg 3.
B. Incorrect because this is not the preferred order as specified in ND-95.5-LB-2 Pg 3.
C. Correct because this is the preferred order as specified in ND-95.5-LP-2 Pg 3.
D. Incorrect because this is not the preferred order as specified in ND-95.5-LP-2 Pg 3

G2.4.38: Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.
Answer: C

95. WEOIG2.1.20 1

Given the following plant conditions following an automatic reactor trip:
- RCS has been verified to be intact per I-E-0, Reactor Trip or Safety Injection
- AFW Flow to "A" SG = 125 gpm
- AFW Flow to "B" SG = I'I0 gpm
- AFW Flow to "C" SG = 130 gprn
- NR "A" SG Level = 10%
- NR "B" SG bevel = *8%*
- NW "C" SG Level = 9%
- RCS Pressure = 1750 psig and slowiy rising
- PRZR Level = **24%** and slowly rising
- RCS subcooling based on CETCs is 80°F

Operators have reached the point in 2-E-0 where they are to check if SI flow should be reduced.

Which ONE of the following would be the next series of operator actions?

A. Direct STA to begin monitoring Critical Safety Function Status Trees, Reset SI and CLS, verify Instrument Air available, then stop all but one Charging Pump, followed by isolating High Head SI **lo** the Cold Legs.

B. Transition to 1-ES-1.1, SI Termination, establish letdown, followed by raising Pressurizer level to > 35%, then secure all but one Charging Pump.

C. Establish letdown, followed **by** raising Pressurizer level to > 35%, transition to **1-E§-'I**.1, SI Termination, then secure all but one Charging Pump.

D. Direct STA to begin monitoring Critical Safety Function Status Trees, Reset SI and **CLS,** verify Instrument Air available, align Charging Pump suction to the VCT, then stop all but one Charging Pump.

Surry

References:
1-E-0, Reactor Trip or Safety Injection, Rev. **46**
I-ES-1.1, SI Termination, Rev. 29
ND-95-03-03, E-0, Reactor Trip of Safety Injection, Rev. 14

Distractor Analysis:
A. Correct because these actions are directed by I-E-0 Steps 26 through 32.
B. Incorrect because letdown would not be established prior to Pzr L > 35%. Plausible because transition to E§-I.1 is logicai and distractor states that the goal is to get Pzr L > 35%.
C. Incorrect because letdown would not be established prior to Pzr L > 35%. Plausible because transition to ES-1.1 is logical and distractor states that the goal is to get Pzr b > 35%.
D. Incorrect becsuse Charging Pump suction would not be aligned to VCT until after all but one Charging Pump is secured. Plausible because all actions are directed by procedure, except that the order of the suction swap and pump stopping is reversed.

W/E01 Rediagnosis and SI Termination
G2.1.20: Ability to execute procedures.
Answer: A

96. WE03EA2.1 1

Operators are responding to a LQCA outside of containment using I-ECA-1.2, LOCA Outside Containment. The crew efforts to isolate the break are unsuccessful.

Which ONE of the following identifies the procedure ECA-1.2 will direct the operators to in order to cool and depressurize the reactor coolant system?

A. I-E-I: boss *of* Reactor or Secondary Coolant

B. 1-ES-1.2, Post LOCA Cooldown and Depressurization

C. 1-ES-1.3, Transfer to Cold Leg Recirculation

D. 1-ECA-I.1, boss *of* Emergency Coolant Recirculation

Surry

References:
1-E-1, Loss of Reactor or Secondary Coolant, Rev. 21
1-ES-1.2, Post LQCA Cooldown and Depressurization, Rev. 21
1-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 12
1-ECA-11, Loss of Emergency Coolant Recirculation, Rev. 17

Distractor Analysis:
A. Incorrect as stated in Distractor D Analysis. Plausible because there is a Loss of Reactor Coolant in progress.
B. Incorrect as stated in Distractor D Analysis. Plausible because the goal is to cool and depressurize the RCS.
C. Incorrect as stated in Distractor D Analysis. Plausible because this is a normal transition for long term cooling during a LOCA.
D. Correct because Step 2 RNO of ECA-1.2 directs operators to ECA-1.1 if efforts to isolate the leak are not successful.

W/E03 LOCA Cooldown - Depress.
EA2.1: Ability to determine and interpret the foliowing as they apply to the (LOCA Cooldown and Depressurization): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Bank Question TP02301.
Answer: D

97. WE05EA2.1 1

The following conditions exist:
- A manual Safety Injection was initiated due to a Steam Break in Safeguards
- All MSTVs have been closed
- All SG pressures are steadily decreasing
- All SG NR levels are off-scale low and WR levels are steadily decreasing
- Pressurizer level is off-scale low
- Pressurizer pressure is steadily decreasing
- RCS temperature is decreasing uncontrollably
- Adequate Auxiliary Feedwater flow exists

Which ONE of the following is the correct procedure transitions for the event in progress?

A. E-0 to E-2 to ECA-2.1

B. E-0 to E-1 to E-2 to ECA-2.1

C. E-0 to E-I to ECA-2.1

D. E-0 to E-2 to E-1

Surry

References:
I-E-0, Reactor Trip or Safety Injection, Rev. 46
I-E-2, Faulted Steam Generator Isolation, Rev. 9
1-ECA-2.1, Uncontrolled Depressurization of All Steam Generators, Rev. 19

Distractor Analysis:
A. ~~Incorrect~~ Correct because E-0 would be entered upon Rx Trip. Step 21 of E-0 sends the team to E-2. Step 2 of E-2 sends the team to ECA-2.1.
B. Incorrect because E-0 Step 21 directs performance of E-2. E-1 is not directed until E-0 Step 23.
C. Incorrect because E-0 Step 21 directs performance of E-2. E-1 is not directed until E-0 Step 23.
D. Incorrect because E-2 would not be entered until after E-1

WE05 Inadequate Heat Transfer - Loss *of* Secondary Heat Sink
EA2.1: Ability to determine and interpret the following as they apply to the (boss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry ILT Bank Question #1342
Answer, A

Adding "All MSTVs have been closed" to the stem ensures the candidates understand that the steam break is unisolable. If closing the MSTVs IAW E-0 actions isolated the break, a different procedure flowpath would occur.

98. WE10EA2.1 1

During a Natural Circulation Cooldown IAW ES-0.3, Natural Circulation Cooldown with Steam Void in **Rx** Vessel, a steam bubble forms in the vessel head. The STA recommends transition to FR-I.3, Response to Voids in Reactor Vessel, to vent the head.

Which ONE of the following courses of action is appropriate?

A. Initiate FR-1.3 since ES-0.3 assumes FR-I.3 is in effect to eliminate the steam void

B. Initiate SI and go to FR-1.3 to vent the head

C. The NC Cooldown should be stopped and a transition to FR-1.3 should be made.

D. Stay in ES-0.3. Void growth is expected and ES-0.3 provides guidance to control the void growth.

Surry

References:
1-FR-I.3, Reponse To Voids In Reactor Vessel, Rev. 16
I-ES-0.3, Natural Circulation Cooldown With Steam Void in **Rx** Vessel, Rev. **22**

Distractor Analysis:
A. Incorrect because ES-0.3 does not assume that FR-1.3 is being used.
B. Incorrect because SI should not be initiated and there *is* no need to vent the head.
C. Incorrect because ES-0.3 does provide guidance for managing void growth.
D. Correct because ES-0.3 does provide guidance for managing void growth.

W/E10 Natural Circ.
EA2.1: Ability to determine and interpret the following as they apply to the (Natural Circulation With Steam Void in Vessl with / without RVLIS): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry Requal Bank Question **#247**
Answer: D

99. WE13EA2.1 1

During performance of E-3, Steam Generator Tube Rupture, the operating team is directed *to* adjust the SG PORV setpoint on the ruptured SG to 1035 psig. The Reactor Operator observes ruptured SG pressure to be 1070 psig and the PORV cycling.

Which ONE *of* the following is the appropriate course of action end reason for the action?

A. Transition to FR-H.2, "Response to Steam Generator Overpressure" to prevent an overpressure condition in the ruptured SG.

B. Increase feed flow to the ruptured **SG** to stop the release and remain in E-3.

C. Increase the setpoint above 1070 psig to prevent release to the public and transition to ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Receovery.

D. Leave the PORV setpoint at 1035 psig to minimize challenges to the SG Code Safeties and remain in E-3.

References:
1-E-3, Steam Generator Tube Rupture, Rev. 25
NB-95.3-LP-13, E-3 Steam Generator Tube Rupture, Rev. 11

Distractor Analysis:
A. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, FR-H.2 **is** not associated with any Red or Orange paths.
B. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, feeding **a** ruptured SG will not limit the exposure to the public.
C. Incorrect because the correct response is simply to verify that the PORV seats when pressure drops below 1035 psig. Furthermore, this action may challenge the code safeties, which is not desirable.
D. Correct because this is the correct direction in the prucedure and the correct basis for the step.

W/E13 Steam Generator Overpressure
EA2.1: Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry Requal **Exam** Bank Question #324
Answer: D

100. WE15EA2.1 2

The Control Room Operators are performing FR-S.2, Response to Loss of Core Shutdown, in response to a yellow path condition shown on the Critical Safety Function (CSF) status tree.

Which ONE of the following is correct with regard to transitions out of this procedure?

A. The operators must leave this procedure at any step as soon as the boss of Core Shutdown CSF adverse condition has cleared. (Green path established)

B. The operators must leave this procedure before completion and go to FR-H.1, Response to boss of Secondary Heat Sink, if the heat sink CSF status tree indicates a yellow path condition.

C. The operators must leave this procedure before completion and go to FR-C.3, Response to Saturated Core Cooling, if the Core Cooling status tree indicates a yellow path condition.

D. The operators must leave this procedure before completion and go to FR-Z.2, Response to Containment Flooding, if the containment CSF status tree indicates an orange path condition.

Surry

Refernces:
ND-95.3-LP-26, Critical Safety Function Status Trees, Rev. 5

Distractor Analysis:
A. Incorrect because the operator does not have to immediately leave FR if it is not completed.
B. Incorrect because yellow path does not warrant this action.
C. Incorrect because yellow path does not warrant this action.
D. Correct because orange path takes priority.

WE15 Containment Flooding
EA2.1: Ability to determine and interpret the following as they apply to the (Containment Flooding): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Surry ILT Bank Question # 1350
Answer: D

Facility: Surry Date of Exam: 3/4/04 Exam Level: ~~RO~~/SRO

	Item Description	Initials a	Initials b	Initials c
1.	Clean answer sheets copied before grading	MB	N/A	SR
2.	Answer key changes and question deletions justified and documented	MB	N/A	SR
3.	Applicants' scores checked for addition errors (reviewers spot check > 25% of examinations)	MB	N/A	SR
4.	Grading for all borderline cases (80 +/- 2% overall and 70 +/- 4% on the SRO-only) reviewed in detail	MB	N/A	SR
5.	All other failing examinations checked to ensure that grades are justified	N/A MB	N/A	SR
6	Performance on missed questions checked for training deficiencies and wording problems; evaluate validity of questions missed by half or more of the applicants	MB	N/A	SR

	Printed Name / Signature	Bate
a. Grader	Mark A. Bates / Mark A. Bates	03-24-2004
b. Facility Reviewer(*)	N/A	
c. NRC Chief Examiner (*)	Steven D. Rose / Steven D. Rose; Edwin Lea, Jr. / Edwin Lea, Jr.	3/25/04 3/24/04
d. NRC Supervisor (*)	Michael E. Ernstes / Michael E. Ernstes	3/25/04

(*) The facility reviewer's signature is not applicable for examinations graded by the NRC; two independent NRC reviews are required.

Facility: SURRY		Date of Exam: 3/4/04	Exam Level: RO/SRO	
	Item Description	Initials a	Initials b	Initials c
1.	Clean answer sheets copied before grading	MB	N/A	SR
2.	Answer key changes and question deletions justified and documented	MB	N/A	SR
3.	Applicants' scores checked for addition errors (reviewers spot check > 25% of examinations)	MB	N/A	SR
4.	Grading for all borderline cases (80 +/- 2% overall and 70 +/- 4% on the SRO-only) reviewed in detail	MB	N/A	SR
5.	All other failing examinations checked to ensure that grades are justified	N/A MB	N/A	SR
6.	Performance on missed questions checked for training deficiencies and wording problems; evaluate validity of questions missed by half or more of the applicants	MB	N/A	SR

	Printed Name / Signature	Date
a. Grader	Mark A. Bates / Mark A. Bates	03-24-2004
b. Facility Reviewer(*)	N/A	
c. NWC Chief Examiner (*)	STEVEN D. ROSE / Steven D. Rose; Edwin Lea, Jr. / Edwin Lea, Jr.	3/25/04; 3/24/04
d. NRC Supervisor (*)	Michael E. Ernstes / Michael E. Ernstes	3/25/04

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SURRY 2004-301

Task Description		Date Complete
1.	Facility written exam comments or graded exams received and verified complete	3/11/04
2.	Facility written exam comments reviewed and incorporated and NRC grading completed, if necessary	3/24/04
3.	Operating tests graded by NRC examiners	3/19/04
4.	NRC Chief examiner review of written exam and operating test grading completed	3/25/04
5.	Responsible supervisor review completed	3/29/04
6.	Management (licensing official) review completed	4/1/04
7.	License and denial letters mailed	4/1/04
8.	Facility notified of results	4/6/04
9.	Examination report issued (refer to NRC MC 0612)	4/6/04
10.	Reference material returned after final resolution of any appeals	N/A