ES-201 **Examination Outline Quality Checklist** Form ES-201-2

Facility: **Hope Creek - RO Exam** Date of Examination: **11/28/05**

Item	Task Description	Initials a	Initials b*	Initials c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model, in accordance with ES-401.	MB	SD	SD
	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	SD	SD
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	MB	SD	SD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MB	SD	SD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.			
	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, and ensure that 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 that scenarios will not be repeated on subsequent days.			
	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.			
3. W/T	a. Verify that the systems walk-through outline meets the criteria specified on Form ES-301-2: (1) the outline(s) contain(s) the required number of control room and in-plant tasks distributed among the safety functions as specified on the form (2) task repetition from the last two NRC examinations is within the limits specified on the form (3) no tasks are duplicated from the applicants' audit test(s) (4) the number of new or modified tasks meets or exceeds the minimums specified on the form (5) the number of alternate path, low-power, emergency, and RCA tasks meet the criteria on the form.			
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) the tasks are distributed among the topics as specified on the form (2) at least one task is new or significantly modified (3) no more than one task is repeated from the last two NRC licensing examinations			
	c. 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.			
* 4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam sections.			
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.			
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.			
	d. Check for duplication and overlap among exam sections.			
	e. Check the entire exam for balance of coverage.			
	f. Assess whether the exam fits the appropriate job level (RO or SRO).			

	Printed Name/Signature	Date
a. Author	Michael L. Brown / [signature]	6/2/05 *
b. Facility Reviewer (*)		
c. NRC Chief Examiner (#)	Steven Dennis / [signature]	6/2/05
d. NRC Supervisor	R. J. Conte / [signature]	6/2/05

Note: # Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

* THIS QASHEET IS FOR WRITTEN EXAM SAMPLE PLAN ONLY

Facility: **Hope Creek - SRO Only Exam** Date of Examination: **11/28/05**

Item	Task Description	Initials a	Initials b*	Initials c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model, in accordance with ES-401.	MB	SD	SD
	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	SD	SD
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	MB	SD	SD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MB	SD	SD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.		SD	
	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, and ensure that 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 that scenarios will not be repeated on subsequent days.			
	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.			
3. W/T	a. Verify that the systems walk-through outline meets the criteria specified on Form ES-301-2: (1) the outline(s) contain(s) the required number of control room and in-plant tasks distributed among the safety functions as specified on the form (2) task repetition from the last two NRC examinations is within the limits specified on the form (3) no tasks are duplicated from the applicants' audit test(s) (4) the number of new or modified tasks meets or exceeds the minimums specified on the form (5) the number of alternate path, low-power, emergency, and RCA tasks meet the criteria on the form.			
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) the tasks are distributed among the topics as specified on the form (2) at least one task is new or significantly modified (3) no more than one task is repeated from the last two NRC licensing examinations			
	c. 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.			
* 4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam sections.			
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.			
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.			
	d. Check for duplication and overlap among exam sections.			
	e. Check the entire exam for balance of coverage.			
	f. Assess whether the exam fits the appropriate job level (RO or SRO).			

	Printed Name/Signature	Date
a. Author	Michael L. Brown / [signature]	6/2/05
b. Facility Reviewer (*)	~~Steven Dennis~~	
c. NRC Chief Examiner (#)	STEVEN DENNIS / [signature]	6/2/05
d. NRC Supervisor	R. J. Conte / [signature]	6/2/05

Note: # Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

* THIS QA SHEET IS FOR WRITTEN EXAM ~~SAMPLAN~~ SAMPLE PLAN ONLY

Facility: **Hope Creek - SRO Only Exam** Date of Exam: **11/28/05**

Tier	Group	RO K/A Category Points: K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points: A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1				N/A					N/A				4	3	7
	2				N/A					N/A				1	2	3
	Tier Totals				N/A					N/A				5	5	10
2. Plant Systems	1													3	2	5
	2													1	2	3
	Tier Totals													4	4	8

3. Generic Knowledge and Abilities Categories	RO: 1	2	3	4	Total	SRO: 1	2	3	4	Total
						2	2	1	2	7

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. * The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	0	0	0	0	0	0			
295003 Partial or Complete Loss of AC / 6	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295004 Partial or Total Loss of DC Pwr / 6	0	0	0	0	1	0	AA2.04 - Ability to determine and interpret the following as they apply to Partial or Total loss of DC power:(CFR: 41.10 / 43.5 / 45.13) - System Lineups	3.3	1
295005 Main Turbine Generator Trip / 3	0	0	0	0	0	0	K/A Randomly Rejected		
295006 SCRAM / 1	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295016 Control Room Abandonment / 7	0	0	0	0	0	0			
295018 Partial or Total Loss of CCW / 8	0	0	0	0	0	0			
295019 Partial or Total Loss of Inst. Air / 8	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Partial or Total loss of Instrument Air:(CFR: 41.10/43.5/ 45.13) - Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	3.7	1
295021 Loss of Shutdown Cooling / 4	0	0	0	0	0	0			
295023 Refueling Acc / 8	0	0	0	0	0	0			
295024 High Drywell Pressure / 5	0	0	0	0	0	0			
295025 High Reactor Pressure / 3	0	0	0	0	0	0			
295026 Suppression Pool High Water Temp. / 5	0	0	0	0	0	0			
295027 High Containment Temperature / 5	0	0	0	0	0	0			
295028 High Drywell Temperature / 5	0	0	0	0	0	1	EG2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR 45.3)	3.3	1
295030 Low Suppression Pool Wtr Lvl / 5	0	0	0	0	1	0	EA2.01 - Ability to determine and interpret the following as they apply to Low Suppression Pool Water level (CFR:41.10/ 43.5/ 45.13) - Suppression Pool level	4.2	1

295031 Reactor Low Water Level / 2	0	0	0	0	0	0			
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	0	0	0	0	0	0			
295038 High Off-site Release Rate / 9	0	0	0	0	0	0			
600000 Plant Fire On Site / 8	0	0	0	0	1	0	AA2.13 - Ability *to determine and interpret the following* as they apply to Plant Fire On Site: (CFR:41.10/ 43.5/ 45.13) - Need for emergency plant shutdown	3.8	1
K/A Category Totals:					4	3	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	0	0	0	0	0	0			
295007 High Reactor Pressure / 3	0	0	0	0	0	0			
295008 High Reactor Water Level / 2	0	0	0	0	0	0			
295009 Low Reactor Water Level / 2	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Low Reactor Water Level (CFR: 41.10/ 43.5 / 45.13) - Steam flow/ feed flow mismatch	3.7	1
295010 High Drywell Pressure / 5	0	0	0	0	0	1	AG2.4.6 - Knowledge of symptom based EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13)	4.0	1
295011 High Containment Temp / 5	0	0	0	0	0	0			
295012 High Drywell Temperature / 5	0	0	0	0	0	0			
295013 High Suppression Pool Temp. / 5	0	0	0	0	0	0			
295014 Inadvertent Reactivity Addition / 1	0	0	0	0	0	0			
295015 Incomplete SCRAM / 1	0	0	0	0	0	0			
295017 High Off-site Release Rate / 9	0	0	0	0	0	0			
295020 Inadvertent Cont. Isolation / 5 & 7	0	0	0	0	0	0			
295022 Loss of CRD Pumps / 1	0	0	0	0	0	0			
295029 High Suppression Pool Wtr Lvl / 5	0	0	0	0	0	0			
295032 High Secondary Containment Area Temperature / 5	0	0	0	0	0	0			
295033 High Secondary Containment Area Radiation Levels / 9	0	0	0	0	0	0			
295034 Secondary Containment Ventilation High Radiation / 9	0	0	0	0	0	0			
295035 Secondary Containment High Differential Pressure / 5	0	0	0	0	0	1	EG2.4.6 - Knowledge of symptom based EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13)	4.0	1

295036 Secondary Containment High Sump/Area Water Level / 5	0	0	0	0	0	0			
500000 High CTMT Hydrogen Conc. / 5	0	0	0	0	0	0			
K/A Category Point Totals:					1	2	Group Point Total:		3

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 1 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode	0	0	0	0	0	0	0	0	0	0	0			
205000 Shutdown Cooling	0	0	0	0	0	0	0	0	0	0	0			
206000 HPCI	0	0	0	0	0	0	0	0	0	0	1	G2.1.14 - Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5 / 45.12)	3.3	1
207000 Isolation (Emergency) Condenser	0	0	0	0	0	0	0	0	0	0	0			
209001 LPCS	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the LPCS and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Valve closures	3.2	1
209002 HPCS	0	0	0	0	0	0	0	0	0	0	0			
211000 SLC	0	0	0	0	0	0	0	0	0	0	0			
212000 RPS	0	0	0	0	0	0	0	0	0	0	0			
215003 IRM	0	0	0	0	0	0	0	0	0	0	0			
215004 Source Range Monitor	0	0	0	0	0	0	0	0	0	0	0			
215005 APRM / LPRM	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the APRM/ LPRM and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Upscale or downscale trips.	3.7	1
217000 RCIC	0	0	0	0	0	0	0	0	0	0	0			
218000 ADS	0	0	0	0	0	0	0	0	0	0	0			
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	0	0	0	0	0	0			
239002 SRVs	0	0	0	0	0	0	0	0	0	0	0			
259002 Reactor Water Level Control	0	0	0	0	0	0	0	0	0	0	1	G2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)	4.0	1

261000 SGTS	0	0	0	0	0	0	0	0	0	0	0			
262001 AC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
262002 UPS (AC/DC)	0	0	0	0	0	0	0	0	0	0	0			
263000 DC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
264000 EDGs	0	0	0	0	0	0	0	1	0	0	0	A2.08 - Ability to (a) predict the impacts of the following on the EDGs and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Initiation of emergency generator room fire protection system.	3.7	1
300000 Instrument Air	0	0	0	0	0	0	0	0	0	0	0			
400000 Component Cooling Water	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 2 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)	3.3	1
201002 RMCS	0	0	0	0	0	0	0	0	0	0	0			
201003 Control Rod and Drive Mechanism	0	0	0	0	0	0	0	0	0	0	0			
201004 RSCS	0	0	0	0	0	0	0	0	0	0	0			
201005 RCIS	0	0	0	0	0	0	0	0	0	0	0			
201006 RWM	0	0	0	0	0	0	0	0	0	0	0			
202001 Recirculation	0	0	0	0	0	0	0	0	0	0	1	G2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 /45.6)	4.0	1
202002 Recirculation Flow Control	0	0	0	0	0	0	0	0	0	0	0			
204000 RWCU	0	0	0	0	0	0	0	0	0	0	0			
214000 RPIS	0	0	0	0	0	0	0	0	0	0	0			
215001 Traversing In-core Probe	0	0	0	0	0	0	0	0	0	0	0			
215002 RBM	0	0	0	0	0	0	0	0	0	0	0			
216000 Nuclear Boiler Inst.	0	0	0	0	0	0	0	0	0	0	0			
219000 RHR/LPCI: Torus/Pool Cooling Mode	0	0	0	0	0	0	0	0	0	0	0			
223001 Primary CTMT and Aux.	0	0	0	0	0	0	0	0	0	0	0			
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
230000 RHR/LPCI: Torus/Pool Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
233000 Fuel Pool Cooling/Cleanup	0	0	0	0	0	0	0	0	0	0	0			
234000 Fuel Handling Equipment	0	0	0	0	0	0	0	0	0	0	0			
239001 Main and Reheat Steam	0	0	0	0	0	0	0	0	0	0	0			
239003 MSIV Leakage Control	0	0	0	0	0	0	0	0	0	0	0			
241000 Reactor/Turbine Pressure Regulator	0	0	0	0	0	0	0	0	0	0	0			

245000 Main Turbine Gen. / Aux.	0	0	0	0	0	0	0	1	0	0	0	A2.05 - Ability to (a) predict the impacts of the following on the Main Turbine Gen. / Aux and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Generator trip	3.8	1
256000 Reactor Condensate	0	0	0	0	0	0	0	0	0	0	0			
259001 Reactor Feedwater	0	0	0	0	0	0	0	0	0	0	0			
268000 Radwaste	0	0	0	0	0	0	0	0	0	0	0			
271000 Offgas	0	0	0	0	0	0	0	0	0	0	0			
272000 Radiation Monitoring	0	0	0	0	0	0	0	0	0	0	0			
286000 Fire Protection	0	0	0	0	0	0	0	0	0	0	0			
288000 Plant Ventilation	0	0	0	0	0	0	0	0	0	0	0			
290001 Secondary CTMT	0	0	0	0	0	0	0	0	0	0	0			
290003 Control Room HVAC	0	0	0	0	0	0	0	0	0	0	0			
290002 Reactor Vessel Internals	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								1			2	Group Point Total:		3

Facility: Hope Creek	Date of Examination: 11/28/05
Examination Level : RO	Operating Test Number:

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	S, A, N	Check Drywell to Torus D/P during power operations per Daily Surveillance Log
Conduct of Operations	R, N	Procedure Change - Make a change to a procedure for Emergent work
Equipment Control	D, S, A	Rod Worth Minimizer Operability -
Radiation Control	R, N, A	Enter and exit a High Radiation Area for a valve lineup. 2.3.10
Emergency Plan		

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

* Type Codes & Criteria: (C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1), (A)lternate Path
(P)revious 2 exams (≤ 1; randomly selected)

Facility: **Hope Creek - SRO Only Exam** Date of Exam: **11/28/05**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)			4.4	1
	2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits (CFR: 41.10 / 43.5 / 45.12)			2.9	1
	Subtotal					2
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities (CFR: 43.5 / 45.13)			3.3	1
	2.2.21	Knowledge of pre- and post-maintenance operability requirements (CFR: 43.2)			3.5	1
	Subtotal					2
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)			3.1	1
	Subtotal					1
4. Emergency Procedures/ Plan	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 43.5 / 45.12)			4.0	1
	2.4.36	Knowledge of chemistry/health physics tasks during emergency operations (CFR: 43.5)			2.8	1
	2.4.					
	Subtotal					2
Tier 3 Point Total						7

Facility: Hope Creek - **RO Exam** Date of Exam: **11/28/05**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.21	Ability to obtain and verify controlled procedure copy (CFR: 45.10 / 45.13)	3.1	1		
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel (CFR: 43.5 / 45.12)	2.5	1		
	2.1.33	Ability to recognize indications for system operating parameters which are entry-level condition for Technical Specifications (CFR: 43.2 / 43.3 / 45.3)	3.4	1		
	2.1.					
	2.1.					
	Subtotal			3		
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1)	3.7	1		
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity (CFR: 43.6)	2.8	1		
	2.2.					
	Subtotal			2		
3. Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements (CFR: 41.12 / 43.4. 45.9 / 45.10).	2.6	1		
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure (CFR: 43.4 / 45.10)	2.9	1		
	2.3.					
	Subtotal			2		

4. Emergency Procedures/ Plan	2.4.27	Knowledge of fire in the plant procedure (CFR: 41.10 / 43.5 / 45.13)	3.0	1		
	2.4.39	Knowledge of the RO's responsibilities in emergency plan implementation (CFR: 45.11)	3.3	1		
	2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)	3.3	1		
	2.4.					
	Subtotal			3		
Tier 3 Point Total				10		

Facility: **Hope Creek - SRO Only Exam** Date of Exam: **11/28/05**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1													4	3	7
	2				N/A					N/A				1	2	3
	Tier Totals													5	5	10
2. Plant Systems	1													3	2	5
	2													1	2	3
	Tier Totals													4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4			1 / 2	3 / 4	7
														2 / 2	1 / 2	

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

e. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7. * The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)								Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	0	0	0	0	0	0			
295003 Partial or Complete Loss of AC / 6	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295004 Partial or Total Loss of DC Pwr / 6	0	0	0	0	1	0	AA2.04 - Ability to determine and interpret the following as they apply to Partial or Total loss of DC power:(CFR: 41.10 J 43.5 / 45.13) - System Lineups	3.3	1
295005 Main Turbine Generator Trip / 3	0	0	0	0	0	0	K/A Randomly Rejected		
295006 SCRAM / 1	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295016 Control Room Abandonment / 7	0	0	0	0	0	0			
295018 Partial or Total Loss of CCW / 8	0	0	0	0	0	0			
295019 Partial or Total Loss of Inst. Air / 8	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Partial or Total loss of Instrument Air:(CFR: 41.10/43,5/ 45.13) - Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	3.7	1
295021 Loss of Shutdown Cooling / 4	0	0	0	0	0	0			
295023 Refueling Acc / 8	0	0	0	0	0	0			
295024 High Drywell Pressure / 5	0	0	0	0	0	0			
295025 High Reactor Pressure / 3	0	0	0	0	0	0			
295026 Suppression Pool High Water Temp. / 5	0	0	0	0	0	0			
295027 High Containment Temperature / 5	0	0	0	0	0	0			
295028 High Drywell Temperature / 5	0	0	0	0	0	1	EG2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR 45.3)	3.3	1
295030 Low Suppression Pool Wtr Lvl / 5	0	0	0	0	1	0	EA2.01 - Ability to determine and interpret the following as they apply to Low Suppression Pool Water level (CFR:41.10/ 43.5/ 45.13) - Suppression Pool level	4.2	1

295031 Reactor Low Water Level / 2	0	0	0	0	0	0			
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	0	0	0	0	0	0			
295038 High Off-site Release Rate / 9	0	0	0	0	0	0			
600000 Plant Fire On Site / 8	0	0	0	0	1	0	AA2.13 - Ability to determine and interpret the following as they apply to Plant Fire On Site: (CFR:41.10/ 43.5/ 45.13) - Need for emergency plant shutdown	3.8	1
K/A Category Totals:					4	3	Group Point Total:		7

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)	Form ES-401-1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	0	0	0	0	0	0			
295007 High Reactor Pressure / 3	0	0	0	0	0	0			
295008 High Reactor Water Level / 2	0	0	0	0	0	0			
295009 Low Reactor Water Level / 2	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Low Reactor Water Level (CFR: 41.10/ 43.5 / 45.13) - Steam flow/ feed flow mismatch	3.7	1
295010 High Drywell Pressure / 5	0	0	0	0	0	1	AG2.4.6 - Knowledge of symptom based EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13)	4.0	1
295011 High Containment Temp / 5	0	0	0	0	0	0			
295012 High Drywell Temperature / 5	0	0	0	0	0	0			
295013 High Suppression Pool Temp. / 5	0	0	0	0	0	0			
295014 Inadvertent Reactivity Addition / 1	0	0	0	0	0	0			
295015 Incomplete SCRAM / 1	0	0	0	0	0	0			
295017 High Off-site Release Rate / 9	0	0	0	0	0	0			
295020 Inadvertent Cont. Isolation / 5 & 7	0	0	0	0	0	0			
295022 Loss of CRD Pumps / 1	0	0	0	0	0	0			
295029 High Suppression Pool Wtr Lvl / 5	0	0	0	0	0	0			
295032 High Secondary Containment Area Temperature / 5	0	0	0	0	0	0			
295033 High Secondary Containment Area Radiation Levels / 9	0	0	0	0	0	0			
295034 Secondary Containment Ventilation High Radiation / 9	0	0	0	0	0	0			
295035 Secondary Containment High Differential Pressure / 5	0	0	0	0	0	1	EG2.4.6 - Knowledge of symptom based EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13)	4.0	1

295036 Secondary Containment High Sump/Area Water Level / 5	0	0	0	0	0	0			
500000 High CTMT Hydrogen Conc. / 5	0	0	0	0	0	0			
K/A Category Point Totals:					1	2	Group Point Total:		3

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 1 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode	0	0	0	0	0	0	0	0	0	0	0			
205000 Shutdown Cooling	0	0	0	0	0	0	0	0	0	0	0			
206000 HPCI	0	0	0	0	0	0	0	0	0	0	1	G2.1.14 - Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5 / 45.12)	3.3	1
207000 Isolation (Emergency) Condenser	0	0	0	0	0	0	0	0	0	0	0			
209001 LPCS	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the LPCS and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Valve closures	3.2	1
209002 HPCS	0	0	0	0	0	0	0	0	0	0	0			
211000 SLC	0	0	0	0	0	0	0	0	0	0	0			
212000 RPS	0	0	0	0	0	0	0	0	0	0	0			
215003 IRM	0	0	0	0	0	0	0	0	0	0	0			
215004 Source Range Monitor	0	0	0	0	0	0	0	0	0	0	0			
215005 APRM / LPRM	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the APRM/ LPRM and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Upscale or downscale trips.	3.7	1
217000 RCIC	0	0	0	0	0	0	0	0	0	0	0			
218000 ADS	0	0	0	0	0	0	0	0	0	0	0			
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	0	0	0	0	0	0			
239002 SRVs	0	0	0	0	0	0	0	0	0	0	0			
259002 Reactor Water Level Control	0	0	0	0	0	0	0	0	0	0	1	G2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)	4.0	1

261000 SGTS	0	0	0	0	0	0	0	0	0	0	0			
262001 AC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
262002 UPS (AC/DC)	0	0	0	0	0	0	0	0	0	0	0			
263000 DC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
264000 EDGs	0	0	0	0	0	0	0	1	0	0	0	A2.08 - Ability to (a) predict the impacts of the following on the EDGs and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Initiation of emergency generator room fire protection system.	3.7	1
300000 Instrument Air	0	0	0	0	0	0	0	0	0	0	0			
400000 Component Cooling Water	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401	BWR Examination Outline Plant Systems - Tier 2/Group 2 (RO / SRO)													Form ES-401-1
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)	3.3	1
201002 RMCS	0	0	0	0	0	0	0	0	0	0	0			
201003 Control Rod and Drive Mechanism	0	0	0	0	0	0	0	0	0	0	0			
201004 RSCS	0	0	0	0	0	0	0	0	0	0	0			
201005 RCIS	0	0	0	0	0	0	0	0	0	0	0			
201006 RWM	0	0	0	0	0	0	0	0	0	0	0			
202001 Recirculation	0	0	0	0	0	0	0	0	0	0	1	G2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 /45.6)	4.0	1
202002 Recirculation Flow Control	0	0	0	0	0	0	0	0	0	0	0			
204000 RWCU	0	0	0	0	0	0	0	0	0	0	0			
214000 RPIS	0	0	0	0	0	0	0	0	0	0	0			
215001 Traversing In-core Probe	0	0	0	0	0	0	0	0	0	0	0			
215002 RBM	0	0	0	0	0	0	0	0	0	0	0			
216000 Nuclear Boiler Inst.	0	0	0	0	0	0	0	0	0	0	0			
219000 RHR/LPCI: Torus/Pool Cooling Mode	0	0	0	0	0	0	0	0	0	0	0			
223001 Primary CTMT and Aux.	0	0	0	0	0	0	0	0	0	0	0			
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
230000 RHR/LPCI: Torus/Pool Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
233000 Fuel Pool Cooling/Cleanup	0	0	0	0	0	0	0	0	0	0	0			
234000 Fuel Handling Equipment	0	0	0	0	0	0	0	0	0	0	0			
239001 Main and Reheat Steam	0	0	0	0	0	0	0	0	0	0	0			
239003 MSIV Leakage Control	0	0	0	0	0	0	0	0	0	0	0			
241000 Reactor/Turbine Pressure Regulator	0	0	0	0	0	0	0	0	0	0	0			

245000 Main Turbine Gen. / Aux.	0	0	0	0	0	0	0	1	0	0	0	A2.05 - Ability to (a) predict the impacts of the following on the Main Turbine Gen. / Aux and (b) based on those predictions, *use procedures to correct* control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Generator trip	3.8	1
256000 Reactor Condensate	0	0	0	0	0	0	0	0	0	0	0			
259001 Reactor Feedwater	0	0	0	0	0	0	0	0	0	0	0			
268000 Radwaste	0	0	0	0	0	0	0	0	0	0	0			
271000 Offgas	0	0	0	0	0	0	0	0	0	0	0			
272000 Radiation Monitoring	0	0	0	0	0	0	0	0	0	0	0			
286000 Fire Protection	0	0	0	0	0	0	0	0	0	0	0			
288000 Plant Ventilation	0	0	0	0	0	0	0	0	0	0	0			
290001 Secondary CTMT	0	0	0	0	0	0	0	0	0	0	0			
290003 Control Room HVAC	0	0	0	0	0	0	0	0	0	0	0			
290002 Reactor Vessel Internals	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								1			2	Group Point Total:		3

Facility:	**Hope Creek - RO Exam**													Date of Exam: **11/28/2005**		
Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			4	3	N/A		4	20			
	2	2	2	1				1	1			0	7			
	Tier Totals	5	5	4				5	4			4	27			
2. Plant Systems	1	2	2	3	3	2	3	2	2	2	2	3	26			
	2	2	1	1	1	1	1	1	2	1	1	0	12			
	Tier Totals	4	3	4	4	3	4	3	4	3	3	3	38			

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	3	2	2	3	10					

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

e. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. * The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	1	0	0	0	0	0	AK1.03 - Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of Forced Core Flow Circulation: Thermal Limits :(CFR: 41.8 to 41.10 / 45.3)	3.6	1
295003 Partial or Complete Loss of AC / 6	0	0	0	0	1	0	AA2.05 - Ability to determine and interpret the following as they apply to Partial or Complete Loss of AC : Whether a partial or complete loss of A.C. Power has occurred:(CFR: 41.10 /43.5/ 45.13)	3.9	1
295004 Partial or Total Loss of DC Pwr / 6	0	0	1	0	0	0	AK3.01 - Knowledge of the reasons for the following responses as they apply to Partial or Total Loss of DC Pwr : Load shedding Plant Specific:(CFR: 41.5/41.10 / 45.6 /45.13)	2.6	1
295005 Main Turbine Generator Trip / 3	0	0	0	0	0	1	AG2.1.2 - Knowledge of operator responsibilities during all modes of plant operation (CFR: 41.10 / 45.13)	3.0	1
295006 SCRAM / 1	1	0	0	0	0	0	AK1.03 - Knowledge of the operational implications of the following concepts as they apply to the SCRAM: Reactivity Control:(CFR: 41.8 to 41.10 /45.3)	3.7	1
295016 Control Room Abandonment / 7	0	0	0	0	0	1	AG2.1.30 - Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	3.9	1
295018 Partial or Total Loss of CCW / 8	0	0	0	0	1	0	AA2.04 - Ability to determine and interpret the following as they apply to Partial or Total Loss of CCW System Flow:(CFR: 41.10/43.5/ 45.13)	2.9	1
295019 Partial or Total Loss of Inst. Air / 8	0	0	0	1	0	0	AA1.03 - Ability to operate and / or monitor the following as they apply to Partial or Total Loss of Inst. Air: Instrument Air Compressor Power supplies:(CFR: 41.7145.5/45.6)	3.0	1
295021 Loss of Shutdown Cooling / 4	0	0	0	0	1	0	AA2.05 - Ability to determine and interpret the following as they apply to Loss of Shutdown Cooling: Reactor Vessel Metal Temperature (CFR: 41.10 /43.5/45.13)	3.4	1
295023 Refueling Acc / 8	0	1	0	0	0	0	AK2.03 - Knowledge of the interrelations between Refueling Accidents and the following: Radiation Monitoring equipment (CFR41.7 /45.7/ 45.8)	3.4	1
295024 High Drywell Pressure / 5	0	0	0	1	0	0	EA1.03 - Ability to operate and/ or monitor the following as they apply to High Drywell Pressure: LPCS - Plant specific (CFR41.7/ 45.5/ 45.6)	4.0	1
295025 High Reactor Pressure / 3	0	0	0	1	0	0	EA1.02 - Ability to operate and / or monitor the following as they apply to High Reactor Pressure : Reactor/Turbine pressure regulating system :(CFR: 41.7/45.5/ 45.6)	3.8	1

295026 Suppression Pool High Water Temp. / 5	0	0	0	0	0	1	EG2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 / 45.6)	3.9	1
295027 High Containment Temperature / 5	0	0	0	0	0	0			0
295028 High Drywell Temperature / 5	0	0	0	0	0	1	EG2.1.30 - Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	3.9	1
295030 Low Suppression Pool Wtr Lvl / 5	0	0	1	0	0	0	EK3.07 - Knowledge of the reasons for the following responses as they apply to Low Suppression Pool Wtr Lvl: NPSH considerations for ECCS pumps:(CFR: 41.5/41.10/45.6/ 45.13)	3.5	1
295031 Reactor Low Water Level / 2	0	1	0	0	0	0	EK2.10 - Knowledge of the interrelations between Reactor Low Water Level and the following: Redundant reactivity control: Plant specific (CFR: 41.7/45.7/45.8)	4.0	1
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	0	0	0	1	0	0	EA1.02 - Ability to operate and / or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: RRCS: Plant Specific (CFR: 41.7/45.5/ 45.6)	3.8	1
295038 High Off-site Release Rate / 9	0	0	1	0	0	0	EK3.02 - Knowledge of the reasons for the following responses as they apply to High Off-site Release Rate: System Isolations :(CFR: 41.5/41.10/45.6/ 45.13)	3.9	1
600000 Plant Fire On Site / 8	1	0	0	0	0	0	AK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Plant Fire On Site: Fire Classifications by type (CFR: 41.8 to 41.10 /45.3) ,	2.5	1
295005 Main Turbine Generator Trip / 3	0	1	0	0	0	0	AK2.04 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection (CFR: 41.7 / 45.8)	3.3	1
K/A Category Totals:	3	3	3	4	3	4	Group Point Total:		20

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Loss of Main Condenser Vacuum's Reactor Power - Plant Specific:(CFR: 41.10/43.5/ 45.13)	3.2	1
295007 High Reactor Pressure / 3	0	0	0	0	0	0			
295008 High Reactor Water Level / 2	0	0	1	0	0	0	AK3.06 - Knowledge of the reasons for the following responses as they apply to High Reactor Water Level: RCIC Turbine Trip - Plant Specific:(CFR: 41.5/41.10/ 45.6/45.13)	3.4	1
295009 Low Reactor Water Level / 2	1	0	0	0	0	0	AK1.02 - Knowledge of the operational implications of the following concepts as they apply to the Low Reactor Water Level: Recirculation pump net positive suction head: Plant Specific:(CFR: 41.8 to 41.10/45.3)	3.0	1
295010 High Drywell Pressure / 5	0	0	0	0	0	0			
295011 High Containment Temp / 5	0	0	0	0	0	0			
295012 High Drywell Temperature / 5	0	0	0	0	0	0			
295013 High Suppression Pool Temp. / 5	0	0	0	0	0	0			
295014 Inadvertent Reactivity Addition / 1	0	0	0	0	0	0			
295015 Incomplete SCRAM / 1	0	0	0	0	0	0			
295017 High Off-site Release Rate / 9	0	0	0	0	0	0			
295020 Inadvertent Cont. Isolation / 5 & 7	0	0	0	0	0	0			
295022 Loss of CRD Pumps / 1	0	0	0	0	0	0			
295029 High Suppression Pool Wtr Lvl / 5	0	1	0	0	0	0	EK2.07 - Knowledge of the interrelations High Suppression Pool Wtr Lvl and the following: Drywell/ containment water level:(CFR: 41.7 /45.7/45.8)	3.1	1
295032 High Secondary Containment Area Temperature / 5	0	0	0	0	0	0			
295033 High Secondary Containment Area Radiation Levels / 9	0	0	0	0	0	0			

295034 Secondary Containment Ventilation High Radiation / 9	0	0	0	1	0	0	EA1.01 - Ability to operate and/ or monitor the following as they apply to Secondary Containment Ventilation High Radiation: Area radiation monitoring system:(CFR41.7/45.5/45.6)	3.8	1
295035 Secondary Containment High Differential Pressure / 5	0	0	0	0	0	0			0
295036 Secondary Containment High Sump/Area Water Level / 5	1	0	0	0	0	0	EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Secondary Containment High Sump/ Area Water Level: Radiation releases (CFR:41.8 to 41.10/45.3)	2.9	1
500000 High CTMT Hydrogen Conc. / 5	0	1	0	0	0	0	EK2.02 - Knowledge of the interrelations between High CTMT Hydrogen Conc. And the following: Containment oxygen monitoring systems (CFR: 41.7 / 45.7 /45.8)	3.1	1
K/A Category Point Totals:	2	2	1	1	1	0	Group Point Total:		7

ES-401												BWR Examination Outline Plant Systems - Tier 2/Group 1 (RO / SRO)		Form ES-401-1
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode	0	0	0	0	0	0	0	0	0	0	1	G2.2.25 - Knowledge of bases in technical specifications for limiting conditions for operations and safely limits (CFR: 43.2)	2.5	1
205000 Shutdown Cooling	0	0	0	0	0	0	0	0	1	0	0	A3.03 - Ability to monitor automatic operations of the Shutdown Cooling System(RHR Shutdown Cooling Mode) including: lights and alarms (CFR:41.7/45.5)	3.5	1
206000 HPCI	0	0	0	0	0	0	0	1	0	0	0	A2.07 - Ability to (a) predict the impacts of the following on the HPCI and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Low suppression pool level: BWR-2, 3, 4 (CFR:41.5/43.5/45.3/45.13)	3.4	1
206000 HPCI	0	0	0	0	1	0	0	0	0	0	0	K5.05 - Knowledge of the operational implications of the following concepts as they apply to the HPCI: Turbine speed control: BWR- 2,3,4 (CFR:41.5/ 45.7)	3.3	1
207000 Isolation (Emergency) Condenser	0	0	0	0	0	0	0	0	0	0	0			
209001 LPCS	0	1	0	0	0	0	0	0	0	0	0	K2.01 - Knowledge of electrical power supplies to the following: Pump power (CFR41.7)	3.0	1
209002 HPCS	0	0	0	0	0	0	0	0	0	0	0			
211000 SLC	0	0	0	1	0	0	0	0	0	0	0	K4.04 - Knowledge of SLC design feature(s) and or interlock(s) which provide for the following: Indication of fault in explosive valve firing circuits (CFR41.7)	3.8	1
212000 RPS	0	0	1	0	0	0	0	0	0	0	0	K3.11 - Knowledge of the effect that a loss or malfunction of the RPS will have on the following: Recirculation system (CFR41.7/45.6)	3.0	1
215003 IRM	0	0	0	1	0	0	0	0	0	0	0	K4.04 - Knowledge of the IRM design feature(s) and or interlock(s) which provide for the following: Varying system sensitivity levels using range switches (CFR41.7)	2.9	1
215003 IRM	0	1	0	0	0	0	0	0	0	0	0	K2.01 - Knowledge of electrical power supplies to the following: IRM Channels/ detectors (CFR41.7)	2.5	1
215004 Source Range Monitor	1	0	0	0	0	0	0	0	0	0	0	K1.02- Knowledge of the physical connections and/or cause-effect relationships between Source Range Monitor and the following: Reactor Manual Control (CFR:41.2 to 41.9/45.7 to 45.8)	3.4	1

215005 APRM / LPRM	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purposes and function of major system components and controls (CFR: 41.7)	3.2	1
217000 RCIC	1	0	0	0	0	0	0	0	0	0	0	K1.05 - Knowledge of the physical connections and/or cause-effect relationships between RCIC and the following: Residual Heat Removal System (CFR:41.2 to 41.9/ 45.7 to 45.8)	2.6	1
218000 ADS	0	0	0	0	0	0	0	1	0	0	0	A2.04 - Ability to (a) predict the impacts of the following on the ADS and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal operation: ADS failure to initiate (CFR 41.5/43.5/45.3/45.13)	4.1	1
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	0	0	0	0	1	0	A4.02 - Ability to manually operate and/or monitor in the control room: Manually initiate the system (CFR:41.7/45.5 to 45.8)	3.9	1
239002 SRVs	0	0	0	0	0	0	0	0	0	1	0	A4.06- Ability to manually operate and/or monitor in the control room: Reactor water level (CFR: 41.7/45.5 to 45.8)	3.9	1
259002 Reactor Water Level Control	0	0	1	0	0	0	0	0	0	0	0	K3.06 - Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control will have on the following: Main Turbine (CFR:41.7/45.6)	2.8	1
261000 SGTS	0	0	1	0	0	0	0	0	0	0	0	K3.02 - Knowledge of the effect that a loss or malfunction of the SGTS will have on the following: Off-site release rate (CFR:41.7/45.6)	3.6	1
262001 AC Electrical Distribution	0	0	0	1	0	0	0	0	0	0	0	K4.03 - Knowledge of AC Electrical distribution design feature(s) and or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers (CFR:41.7)	3.1	1
262002 UPS (AC/DC)	0	0	0	0	0	1	0	0	0	0	0	K6.02 - Knowledge of the effect that a loss or malfunction of the following will have on the UPS (AC/DC): DC electrical power (CFR:41.7/45.7)	2.8	1
263000 DC Electrical Distribution	0	0	0	0	0	0	1	0	0	0	0	A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical distribution controls including: Battery charging/discharging rate (CFR:41.5/45.5)	2.5	1
264000 EDGs	0	0	0	0	0	0	0	0	0	0	1	G2.1.14 - Knowledge of system status criteria which require notification of plant personnel (CFR: 43.5 / 45.12)	2.5	1
300000 Instrument Air	0	0	0	0	0	0	0	0	1	0	0	A3.02 - Ability to monitor automatic operations of the Instrument Air including: Air temperature (CFR 41.7/45.5)	2.9	1
300000 Instrument Air	0	0	0	0	1	0	0	0	0	0	0	K5.01 - Knowledge of the operational implications of the following concepts as they apply to the Instrument Air: Air Compressors (CFR:41.5/ 45.7)	2.5	1
400000 Component Cooling Water	0	0	0	0	0	1	0	0	0	0	0	K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the Component Cooling Water: Valves (CFR:41.5/45.5)	2.7	1

215004 Source Range Monitor	0	0	0	0	0	0	1	0	0	0	0	A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: RPS status (CFR: 41.5 / 45.5)	3.4	1
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	1	0	0	0	0	0	K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF :Nuclear boiler instrumentation (CFR: 41.7 / 45.7)	3.3	1
K/A Category Point Totals:	2	2	3	3	2	3	2	2	2	2	3	Group Point Total:		26

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 2 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic	0	0	0	0	0	0	0	0	0	0	0			
201002 RMCS	0	0	0	0	0	0	0	0	0	0	0			
201003 Control Rod and Drive Mechanism	0	0	0	0	0	0	0	0	0	0	0			
201004 RSCS	0	0	0	0	0	0	0	0	0	0	0			
201005 RCIS	0	0	0	0	0	0	0	0	0	0	0			
201006 RWM	0	0	0	0	0	1	0	0	0	0	0	K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the RWM: Rod Position indication - Plant Specific	2.9	1
202001 Recirculation	0	0	0	0	0	0	0	0	0	0	0			
202002 Recirculation Flow Control	0	0	0	0	0	0	0	1	0	0	0	A2.07 - Ability to (a) predict the impacts of the following on the Recirculation flow control and (b) based on those predications, use procedures to correct, control, or mitigate the conseqences of those abnormal operation: Loss of feedwater singal inputs: Plant specific (CFR:41.5/43.5/45.3/45.13)	3.3	1
204000 RWCU	0	0	0	0	0	0	0	0	0	0	0			
214000 RPIS	0	0	0	0	0	0	0	0	0	0	0			
215001 Traversing In-core Probe	0	0	0	0	0	0	0	0	0	0	0			
215002 RBM	0	0	0	0	0	0	0	0	0	0	0			
216000 Nuclear Boiler Inst.	0	0	0	0	0	0	0	0	0	0	0			
219000 RHR/LPCI: Torus/Pool Cooling Mode	0	0	0	1	0	0	0	0	0	0	0	K4.03 - Knowledge of RHR/LPCI Torus/Pool Cooling Mode design feature(s) and or interlocks which provide for the following: Unintentional reduction in vessel injection flow during accident conditions: plant specific (CFR:41.7)	3.8	1
223001 Primary CTMT and Aux.	0	0	0	0	0	0	0	0	0	0	0			
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
230000 RHR/LPCI: Torus/Pool Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
233000 Fuel Pool Cooling/Cleanup	0	0	0	0	0	0	0	0	0	0	0			
234000 Fuel Handling Equipment	0	0	0	0	0	0	0	0	0	0	0			
239001 Main and Reheat Steam	0	0	0	0	0	0	0	0	1	0	0	A3.01 - Ability to monitor automatic operations of the Main and Reheat system including: Isolation of main steam system (CFR:41.7/45.5)	4.2	1

239003 MSIV Leakage Control	0	0	0	0	0	0	0	0	0	0	0			
241000 Reactor/Turbine Pressure Regulator	0	0	0	0	0	0	0	0	0	0	0			
245000 Main Turbine Gen. / Aux.	1	0	0	0	0	0	0	0	0	0	0	K1.02 - Knowledge of the physical connections and/or cause effect relationships between Main Turbine Generator / Aux and the following: Condensate system (CFR:41.2 to 41.9 / 45.7 to 45.8)	2.5	1
256000 Reactor Condensate	0	0	0	0	0	0	0	0	0	0	0			
259001 Reactor Feedwater	0	0	0	0	0	0	0	0	0	0	0			
268000 Radwaste	0	0	0	0	0	0	0	1	0	0	0	A2.01 - Ability to (a) predict the impacts of the following on the Radwaste and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal operation: System rupture (CFR:41.5/ 43.5/ 45.3/ 45.13)	2.9	1
271000 Offgas	0	0	0	0	0	0	0	0	0	0	0			
272000 Radiation Monitoring	0	0	0	0	1	0	0	0	0	0	0	K5.01 - Knowledge of the operational implications of the following concepts as they apply to the Radiation Monitoring: Hydrogen injection operation's effect on process radiation indications: Plant specific (CFR: 41.5/ 45.7)	3.2	1
286000 Fire Protection	0	1	0	0	0	0	0	0	0	0	0	K2.03 - Knowledge of electrical power supplies to the following: *Fire detection system: Plant specific* (CFR:41.7)	2.5	1
288000 Plant Ventilation	0	0	0	0	0	0	0	0	0	0	0			
290001 Secondary CTMT	0	0	0	0	0	0	0	0	0	0	0			
290003 Control Room HVAC	0	0	0	0	0	0	1	0	0	0	0	A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the Control Room HVAC controls including: Area Temperatures (CFR41.5/45.5)	2.6	1
290002 Reactor Vessel Internals	1	0	0	0	0	0	0	0	0	0	0	K1.20 - Knowledge of the physical connections and/or cause effect relationships between Reactor Vessel Internals and the following: Nuclear Instrumentation (CFR:41.2 to 41.9/ 45.7 to 45.8)	3.2	1
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	1	0	A4.14 - Ability to manually operate and/or monitor in the control room: Suppression pool temperature *(CFR: 41.7 / 45.5 to 45.8)*	3.5	1
223001 Primary CTMT and Aux.	0	0	1	0	0	0	0	0	0	0	0	K3.01 - Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: Secondary containment (CFR: 41.7 / 45.4)	3.6	1
K/A Category Point Totals:	2	1	1	1	1	1	1	2	1	1	0	Group Point Total:		12

Facility: Hope Creek - **RO Exam**		Date of Exam: **11/28/05**				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.21	Ability to obtain and verify controlled procedure copy (CFR: 45.10 / 45.13)	3.1	1		
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel (CFR: 43.5 / 45.12)	2.5	1		
	2.1.33	Ability to recognize indications for system operating parameters which are entry-level condition for Technical Specifications (CFR: 43.2 / 43.3 / 45.3)	3.4	1		
	2.1.					
	2.1.					
	Subtotal			3		
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1)	3.7	1		
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity (CFR: 43.6)	2.8	1		
	2.2.					
	Subtotal			2		
3. Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements (CFR: 41.12 / 43.4. 45.9 / 45.10).	2.6	1		
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure (CFR: 43.4 / 45.10)	2.9	1		
	2.3.					
	Subtotal			2		

4. Emergency Procedures/ Plan	2.4.27	Knowledge of fire in the plant procedure (CFR: 41.10 / 43.5 / 45.13)	3.0	1		
	2.4.39	Knowledge of the RO's responsibilities in emergency plan implementation (CFR: 45.11)	3.3	1		
	2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)	3.3	1		
	2.4.					
	Subtotal			3		
Tier 3 Point Total				10		

Facility: **Hope Creek - SRO Only Exam** Date of Exam: **11/28/05**

Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)			4.4	1
	2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits (CFR: 41.10 / 43.5 / 45.12)			2.9	1
	Subtotal					2
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities (CFR: 43.5 / 45.13)			3.3	1
	2.2.21	Knowledge of pre- and post-maintenance operability requirements (CFR: 43.2)			3.5	1
	Subtotal					2
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)			3.1	1
	Subtotal					1
4. Emergency Procedures/ Plan	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 43.5 / 45.12)			4.0	1
	2.4.36	Knowledge of chemistry/health physics tasks during emergency operations (CFR: 43.5)			2.8	1
	2.4.					
	Subtotal					2
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
Tier 1/ Group 1 RO exam	295027 EK2.01	K/A is for a Mark III containment and Hope Creek has a Mark I containment
Tier 2/ Group 1 RO exam	259002, A1.06	Hope Creek does not have (FWCI) Feedwater Coolant Injection
Tier 2/ Group 1 RO exam	262002, A1.02	Not applicable to Hope Creek
Tier 2/ Group 2 RO exam	215002, A4.04	Not applicable to Hope Creek
Tier 2/ Group 2 RO exam	223001 A4.02	Not applicable to Hope Creek
Tier 3 RO exam	G2.2.3	Not applicable to Hope Creek - Not a Multi-unit facility

* FIRST 52 WRITTEN QUESTIONS ONLY

ES-401	Written Examination Quality Checklist	Form ES-401-6 *

Facility: Date of Exam: Exam Level: RO ☑ SRO ☑

Item Description					Initial a	Initial b*	Initial c#
1. Questions and answers are technically accurate and applicable to the facility.					MB		SD
2. a. NRC K/As are referenced for all questions. b. Facility learning objectives are referenced as available.					MB		SD
3. SRO questions are appropriate in accordance with Section D.2.d of ES-401					MB		SD
4. The sampling process was random and systematic (if more than 4 RO and 2 SRO questions are repeated from the last 2 NRC licensing exams, consult with NRR OL program office.)							
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		SD
6. Bank use meets limits (no more than 75 percent from the bank, at least 10 percent new, and the rest new or modified); enter the actual RO / SRO-only question distribution(s) at right.	Bank: 19 / 1	Modified: 17 / 4	New: 9 / 2		MB		SD
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: 17 / 1	C/A: 28 / 6			MB		SD
8. References/handouts provided do not give away answers or aid in the elimination of distractors.					MB		SD
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		SD
10. Question psychometric quality and format meet the guidelines in ES Appendix B.					MB		SD
11. The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with the value on the cover sheet.					N/A		SD

* NOTE only 1st 52 Questions

	Printed Name / Signature	Date
a. Author	Michael L. Brown / Michael L. Brown	8/8/05
b. Facility Reviewer (*)		
c. NRC Chief Examiner (#)	Steven Dennis / [signature]	8/11/05
d. NRC Regional Supervisor	R. J. Conte / [signature]	8/12/05

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.

* Draft First 52 Questions

Question	1

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO ☐ SRO

Tier #	1	Group #	1

Importance 3.6

295001 — Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

AK1.03 — Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of Forced Core Flow Circulation — Thermal Limits :(CFR: 41.8 to 41.10 / 45.3)

Question

Given: Hope Creek was at 100% power when the "B" Recirc pump developed excessive vibration and needed to be tripped.

WHICH ONE of the following actions is REQUIRED to be taken in accordance with either HC.OP-AB.RPV-0003(Q), Recirculation System or HC.OP-IO.ZZ-0006, Power Changes during Operation?

A The MAPLHGR limits must be reduced.

B The discharge valve HV-F031B must be closed and maintained closed.

C The MCPR safety limit must be reduced.

D Speed control for the operating pump must be placed in Master Manual Control.

Answer A

References HC.OP-AB.RPV-0003 (Q), Rev. 9, Recirculation System
HC.OP-IO.ZZ-0006,Rev. 33, Power Changes during Operation

Justification

References during Exam None

A. is CORRECT per IOP-6, step. 5.3.7
B. is INCORRECT per RPV-3, Condition A, step A.3 which states that the discharge valve must be closed for approximately 5 minutes and then re-opened .
C. is INCORRECT per IOP-6, step 5.3.7 which states that the MCPR safety limit must be raised
D. is INCORRECT per IOP-6, step 5.3.7 which states that speed control must be placed in Local Manual Control

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:

SXD review - 7/21/05 - LOD 1.75 perhaps re-write to make more difficult, removed "initially" from in front of at 100% power in stem.

Question 2		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 1 *Group #* 1		
☐ *SRO*	*Importance* 3.9		

295003 Partial or Complete Loss of AC / 6

AA2.05 Ability to determine and interpret the following as they apply to Partial or Complete Loss of AC — Whether a partial or complete loss of A.C. Power has occurred:(CFR: 41.10 /43.5/ 45.13)

Question

Given the following conditions:
·The plant is in Operational Condition 5 with the Electrical Distribution System aligned in the Normal lineup.
·An internal short on Transformer 1BX-501 causes a sudden pressure fault on the transformer.

Which one of the following describes the resulting availability of power for the Safe Shutdown Systems?

A Power is lost permanently to both 4.16KV switchgear 10A401 and 10A403. 13 KV breakers BS 2-3 and BS 1-2 stay closed. B and D Diesel Generators start but their output breakers DO NOT CLOSE.

B Power is lost momentarily to both 4.16KV switchgear 10A402 and 10A404. 13 KV breakers BS 2-3 and BS 1-2 trip open. Power is restored when the B and D Diesel generators output breakers close.

C Power is lost momentarily to both 4.16KV switchgear 10A402 and 10A404. 13 KV breakers BS 2-3 and BS 1-2 trip open. Power is restored when the alternate supply breaker from Transformer 1AX501 closes. B and D diesel generators START but their output breakers DO NOT CLOSE.

D Power is lost momentarily to both 4.16KV switchgear 10A402 and 10A404. 13 KV breakers BS 2-3 and BS 1-2 trip open. Power is restored when the alternate supply breaker from Transformer 1AX501 closes. B and D diesel generators DO NOT START.

Answer D

References Hope Creek Question Q76871 - Modified
Drawing E-0001 and 066-01: Class 1E AC Power Distribution
NOH01EAC00-02 - CLASS 1E AC POWER DISTRIBUTION, page 32 of 93

Justification

References during Exam Drawing E-0001

Justification:
Correct answer. 13 Kv Breakers BS 2-3 and BS 1-2 trip open. Bus section 2 is de-energized, Bus section 1 remains energized. The bus infeed breaker swap to the AX501 feed. The loads remain energized. Because one infeed is always available, the Diesels do not start.
A - INCORRECT - Power is not permanently lost to both 4.16KV switchgears. Power is restored when the bus infeed breaker swaps to the AX501 feed.
B - INCORRECT - Power is not restored from the B & D Diesel Generators
C - INCORRECT - The B & D Diesel Generators DO NOT START

Question Source Mod ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:
SXD Review 7/21/05 - OK

Question	3

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO

☐ SRO

Tier # 1 **Group #** 1

Importance 2.6

295004 Partial or Total Loss of DC Pwr / 6

AK3.01 Knowledge of the reasons for the following responses as they apply to Partial or Total Loss of DC Pwr — Load shedding Plant Specific:(CFR: 41.5/41.10 / 45.6 /45.13)

Question

With the plant at 100% power, the plant loses power to 125V DC Class 1E switchgear 10D410.

If the plant were to experience a LOCA, how will Load shedding and control of non-1E loads be affected:

Load shedding of Non-1E loads that get control power from 10D410 ...

A will still occur and these loads can be still be operated from the Control Room (ie. Load shedding and control will not be affected)

B will still occur, however, these loads can NOT be operated from the Control Room.

C will not occur, however, these loads can still be operated from the Control Room.

D will not occur and these loads can NOT be operated from the Control Room.

Answer D

References
INPO Question 23597 (somewhat)
Hope Creek Lesson Plan NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION p34 talks about load shedding of non-1E loads on a LOCA
NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION p.22 talks about 125V DC supplying breaker control power

Justification

References during Exam None

A. - CORRECT - loss of DC control power will result in the loss of all remote control to the affected breaker.
B. - INCRRECT - load will NOT auto trip on a Load Shed Signal and CANNOT be operated from the Control Room.
C. - INCORRECT - load will NOT auto trip on a Load Shed signal
D. - INCORRECT - load CANNOT be operated from the Control Room

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD Review - 7/21 - Question Stem confusing -
7/27 - Rewrote Question Stem - re-submitted
8/2 JD - Weak Question, doesn't address K/A - K/A Q about DC load manual shedding to conserve battery life
8/3 - rewrote question again.

Question 4		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 1	*Group #* 1	
☐ *SRO*	*Importance* 3		

295005 Main Turbine Generator Trip / 3

AG2.1.2 Knowledge of operator responsibilities during all modes of plant operation (CFR: 41.10 / 45.13)

Question

Due to a main turbine vibration problem with a generator load of 110 MWe, a manual turbine trip is performed.

Which of the following describes when the operator is REQUIRED (Maximum Time Limit) to open the generator Output Breakers for the given conditions? (Assume they have not already tripped on reverse power.)

A Immediately

B Within 15 seconds of the turbine trip

C Within 60 seconds of the turbine trip

D Within 90 seconds of the turbine trip

Answer B *References* Hope Creek Question - Q53470
HC.OP-SO.AC-0001(Q) - Rev. 48, MAIN TURBINE OPERATION - P&L 3.1.15

Justification *References during Exam* None

Correct Answer: "15 seconds" -Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power.
The following distractors are incorrect as follows:
"immediately" - Procedure caution calls for operator actions within 15 seconds
"60 seconds" - Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power.
"90 seconds"-Procedure caution calls for operator actions within 15 seconds of the turbine trip at low power. Only when above 150 MWe is the time extended to 90 seconds.

Question Source Bank ☑ *Memory Level* ☐ *Comprehension Level*

Question History:

SXD Review - 7/21 - Had question about lower power -
7/27 - verified power level ok per IOP-4 p.15

Question	5

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3.7

295006 SCRAM / 1

AK1.03 Knowledge of the operational implications of the fallowing concepts as they apply to the SCRAM — Reactivity Control:(CFR: 41.8 to 41.10 /45.3)

Question

A reactor scram has just occurred and the crew is executing HC.OP-AB.ZZ-0000, REACTOR SCRAM.

Which of the following is the reason that step S-8 directs the operator to RESET the scram (SB) if conditions permit AND INSERT a Half-Scram (if Required)?

A To reduce the potential for CRD pump runout and reduce the amount of time for the HCU accumulators to recharge.

B To restore the CRD hydraulic system to normal for insert and withdrawal capability if rods are found at the 02 or beyond position.

C To reestablish the normal primary vessel boundaries by isolating the CRD HCU from the scram discharge volume (SDV) and closing the SDV vent and drain valves.

D To prevent excessive discharge of hot radioactive water to the Reactor Building Equipment Drain Sump.

Answer B **References** Hope Creek Question - Q56128
NOH01AB0000-01, Reactor Scram AB-0000 p.14

Justification **References during Exam** None

Justification:
A - INCORRECT - To reestablish the normal primary vessel boundaries by isolating the CRD HCU from the scram discharge volume (SDV) and closing the SDV vent and drain valves. Incorrect – the Scram reset will open the vents and drains
B - CORRECT - To restore the CRD hydraulic system to normal for insert and withdrawal capability if rods are found at the 02 or beyond position. Correct.
C - INCORRECT -To reduce the potential for CRD pump runout and reduce the amount of time for the HCU accumulators to recharge. Incorrect – system flow restricting orifice limit pump runout to 200 gpm
D - INCORRECT - To prevent excessive discharge of hot radioactive water to the Reactor Building Equipment Drain Sump. Incorrect – resetting scram will send water to the Rx Bldg Equipment Drain Sump

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

Submitted 7/22
SXD Reviewed 7/23 - for Distactor C - asked is this verified?

Question	6	Exam-Cross-Ref
☑ RO	Tier # 1	Group # 1
☐ SRO	Importance 3.9	

295016 Control Room Abandonment / 7

AG2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Question

Remote Shutdown Panel Transfer Switch "B" has been placed in the EMERGENCY position.

Which of the following lists the SRVs that can be operated at the Remote Shutdown Panel (10C399) AND describes the status of their controls in the Control Room (CR)?

A A, B, C, D, & E. CR controls still function normally

B A, B, C, D, & E. CR controls are disabled

C F, H, & M. CR controls still function normally

D F, H & M. CR controls are disabled.

Answer D

References Hope Creek Question - Q62205, HC.OP-IO.ZZ-0008, Section 5.1, Attachment #1, Step B.2.9

NOH01MSTEAMC-02, MAIN STEAM SYSTEM, Obj R3d

Justification

References during Exam None

D - CORRECT - F, H & M. CR controls are disabled. Only SRVs M, F & H can be controlled from the RSP and when the transfer switches are in EMERGENCY, the CR functions are disabled.

A - INCORRECT - A, B, C, D & E are the ADS valves not the valves that can be controlled from the RSP.
B - INCORRECT - A, B, C, D & E CANNOT be controlled from the RSP.
C - INCORRECT - CR controls are disabled when RSP transfer switch B has been placed in the EMERGENCY position.

Question Source Bank ☑ Memory Level ☐ Comprehension Level

Question History:

SXD Review - 7/21 - LOD 1.75 evaluate Revising

Question	7

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 2.9

295018 Partial or Total Loss of CCW / 8

AA2.04 Ability to determine and interpret the following as they apply to Partial or Total Loss of CCW :(CFR: 41.10/43.5/ 45.13) System Flow

Question

With the plant operating at 100% power, power is lost to one of the two Operating SACS pumps due to a breaker fault.

After completing all immediate and subsequent actions of HC.OP-AB.COOL-0002, SAFETY/TURBINE AUXILIARIES COOLING SYSTEM there ____(1)____ flow in both loops of SACS. The most restrictive LCO for this condition requires the plant to be placed in cold shutdown within ____(2)____.

Complete the blank statements from the list below:

A (1)is
(2) 31.5 days (30 days + 12 hours + 72 hours)

B (1)is not
(2)4 days (72 hours + 24 hours)

C (1)is
(2) 4 days (72 hours + 24 hours)

D (1)is not
(2) 31.5 days (30 days + 12 hours + 72 hours)

Answer A

References INPO Question 25996
Hope Creek Procedure HC.OP-AB.COOL-0002, SAFETY/TURBINE AUXILIARIES COOLING SYSTEM, p. 9-13

Justification

References during Exam Tech Specs - 3.7.1.1 -> 3.7.1.3

A - CORRECT - Per AB.COOL-0002, p. 9 when the Operating SACS pump trips, TACS will receive an AUTO SWAP signal, this section of the procedure ensures that the Standy SACS pump auto starts and then also ensures that if the Associated SACS pump tripped due to Low Delta P it is restarted. Thus at the end of the procedure, flow has been restored to both loops. Per TS 3.7.1.1 inop pump must be returned to service within 30 days or Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.
B - INCORRECT - flow will be restored to both loops
C - INCORRECT - TS action must be taken within 31.5 days. This answer is plausible if the student mis-reads TS and determines that either one of the 2 Diesel Generators or Service Water pumps are inoperable per the *** Note.
D - INCORRECT - flow will be restored to both loops.

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD Review 7/21 - Maybe SRO level question, maybe a direct lookup
7/27 - I don't think it's a direct lookup - Look up Lesson Plan Objective

Question	8

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3

295019 Partial or Total Loss of Inst. Air / 8

AA1.03 Ability to operate and/or monitor the following as they apply to Partial or Total Loss of Inst. Air — Instrument Air Compressor Power supplies:(CFR: 41.7145.5/45.6)

Question

Given the following conditions:

Hope Creek is starting up from a Refueling outage, the plant is currently in OPCON 3 with temperature at 240°F and with the Instrument Air pressure at 105 psig and the Instrument//Service Air Systems aligned as follows:

Compressor	Control Mode	Status
00K107	MAN	Running
10K107	MAN	OFF
10K100	AUTO	OFF

A Maintenance Worker accidentally bumps into 7.2KV Bus 10A120 causing it's input breaker to open and the bus to de-energize.

Assuming no operator actions, which of the following correctly states the expected response of the Instrument/ Service Air systems?

A Service Air compressor 10K107 de-energizes, Instrument Air header pressure remains at 105 psig.

B Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 92 psig, when Service Air Compressor 10K107 starts and returns pressure to ~95 psig.

C Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 85 psig when Emergency Air Compressor 10K100 starts and returns pressure to ~105 psig.

D Service Air compressor 00K107 de-energizes, Instrument Air header pressure drops to 85 psig when Emergency Air Compressor 10K100 starts and returns pressure to ~95 psig.

Answer D **References** NOH01SERAIR-01, SERVICE AIR SYSTEM, p.47-48
NOH01INSAIR-01, INSTRUMENT AIR SYSTEM, p15, 42

Justification **References during Exam** None

A. INCORRECT - Power to SAC 10K107 is from 7.2 KV bus 10A110, not 10A120
B. INCORRECT - SAC 10K107 will not start at 92 psig because it's in MAN control.
C. INCORRECT - EIAC 10K100 will auto start at 85 psig, however, it unloads at 100 psig, thereby making in not capable of raise pressure to 105 psig.
D. CORRECT - Loss of Power to 10A120 causes a loss of Power to SAC 00K107, Instrument Air header pressure drops to 85 psig, when EIAC 10K100 starts and brings pressure back to some value < 100 psig.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

SXD reviewed 7/25 - minor editorial changes to stem and distractor B - changed 105 psig to 95 psig.

Question	9

☑ RO
☐ SRO

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

Tier #	1	Group #	1

Importance 3.4

295021 Loss of Shutdown Cooling / 4

AA2.05 Ability to determine and interpret the following as they apply to Loss of Shutdown Cooling — Reactor Vessel Metal Temperature (CFR: 41.10 /43.5/45.13)

Question

Given the following conditions:

·The reactor has been shutdown for 90 hours following 1000 EFPD of operation.
·The plant is in Op Cond 4 with coolant temperature at 140°F.
·A total loss of Shutdown Cooling occurred at 1200 hours.
·All efforts to restore heat removal from the RPV have failed.

Assuming no additional operator action, when will the plant reach OPCON 3?

A 1245

B 1307

C 1330

D 1352

Answer B

References Hope Creek Question - Q61328, HC.OP-AB.RPV-0009, Figure 1and Technical Specification Table 1.2

Justification

References during Exam Figure 1 of HC.OP-AB.RPV-0009

Justification
·1307- correct- Operational Condition 3 is achieved when the Reactor temperature reaches 200°F. The 140°F curve of Figure 1 intersects the 90-hour line between the 1.000 and 1.250 hour lines. 1307 is the only option that is between 1 hour and 1 hour and fifteen minutes following the loss of SDC.
·1245. incorrect- Value obtained by using the 180°F curve.
·1330. -incorrect- Value obtained by using the 120°F curve.
·1352. -incorrect- Value obtained by using the 100°F curve.

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:
SXD Review 7/21 - OK

Question	10

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3.4

295023 Refueling Acc / 8

AK2.03 Knowledge of the interrelations between Refueling Accidents and the following — Radiation Monitoring equipment (CFR41.7 /45.7/ 45.8)

Question

Given the following conditions:

-The plant is in a refueling outage with a fuel move in progress.
-The 'A' Refuel Floor Radiation Monitor has failed downscale. No actions have been taken to address this failure.
-At time 0000 a fuel bundle is dropped and radiation levels on the refuel floor start to slowly rise.
-At time 0005 the B Refuel Floor Radiation Monitor reaches its Hi Trip Setpoint.
-At time 0010 the C Refuel Floor Radiation Monitor reaches its Hi Trip Setpoint.

Under these conditions, an automatic trip of the Reactor Building Ventilation Exhaust (RBVE) fans due to Hi Refuel Floor Radiation levels:

A will occur at time 0010.

B is effectively disabled due to the 'A' Refuel Floor Radiation Monitor being failed downscale

C will occur at time 0005.

D will NOT occur until at least 1 Reactor Building exhaust radiation monitor senses high radiation.

Answer A

References INPO Question 25978
NOH04000221C-01, RADIATION MONITORING SYSTEM p. 29

Justification

References during Exam None

A. _ CORRECT - Per lesson plan p.29 item g. Automatic actions on a Refuel Floor Exhaust RM-23A HIGH radiation intensity level (any two of the three) - RBVE fans trip.
B - INCORRECT - still have 2/3 monitors available
C - INCORRECT - since A channel is failed downscale, need 2/3 to get actuation. Therefore won't get actuation when B channel gets high signal.
D - INCORRECT - will get a trip of RBVE fans on either Hi Refuel Floor Rad levels and RBVE rad levels.

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:

SXD Review 7/21 - Changed Distractor D to make it clearer

Question	11

☑ RO
☐ SRO

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

Tier # 1 **Group #** 1

Importance 4

295024 High Drywell Pressure / 5

EA1.03 Ability to operate and/ or monitor the following as they apply to High Drywell Pressure LPCS

Question

The A Core Spray pump is in full flow test mode in accordance with HC-OP.IS.BE-0001, Core Spray Pumps A and C Inservice Test. A steam leak in the drywell has caused the following conditions:

Reactor was scrammed and all rods inserted.
RPV level reached -60 inches and is now rising with HPCI.
Drywell pressure is 3.0 psig rising.
RPV pressure is 800 psig lowering.
Offsite power remains available to the 4KV buses.

Based on the above conditions, which one of the following is the correct response of the Core Spray system?

A "A" Core Spray pump continues to run in full flow test, all others are operating in min flow.

B ALL Core Spray pumps are operating on min flow.

C ALL Core Spray pumps are tripped and ALL pumps will start when RPV pressure lowers to 461 psig.

D ALL Core Spray pumps are injecting.

Answer B

References INPO Question 24762
NOH01CSSYS0-01, CORE SPRAY SYSTEM

Justification

References during Exam None

A - INCORRECT - Core spray full flow test valve closes upon Receipt of a CSS initiation signal.

B - CORRECT - Core Spray received a start signal at DW pressure > 1.68 psig. This caused all Core Spray pumps to start, however, RPV pressure is > 461 psig so upstream injection valves are closed and pumps are operating on their mini-flow valves. Core Spray test valve auto closed upon receipt of a CSS initiation signal.

C - INCORRECT - Core Spray pumps receive a start signal with pressure > 1.68 psig.

D - INCORRECT - Core Spray pumps upstream injection valves don't open until RPV pressure is < 461 psig.

"initiation" pump start signal is reached., A Core Spray running, no trip signal to any CS pumps and no loss of power., No Core Spray "initiation" pump start signal is reached., Correct, > 2 psig signal closes full flow test valve

Question Source Mod ☐ **Memory Level** ☑ **Comprehension Level**

Question History:

SXD review 7/21 - OK
8/2 JD - Minor editorial change to "A" distractor - Incorporated

Question	13	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 1 Group # 1		
☐ SRO	Importance 3.9		

295026 Suppression Pool High Water Temp. / 5

EG2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 / 45.6)

Question

While implementing HC.OP-EO.ZZ-101, Step RC/L-5, the following plant conditions exist:

- Suppression Pool Temperature is 230°F.
- RPV Water Level is –100 inches and rising.
- RHR Pump AP202 is injecting at 5,000 gpm.
- RHR Pump BP202 is injecting at 10,000 gpm.
- Loop A Core Spray is injecting at 1,000 gpm.
- Loop B Core Spray is injecting at 2,200 gpm.
- Suppression Chamber pressure is 5.0 psig.
- Suppression Chamber water level is 0 inches.

Which one of the following describes an action to be taken to ensure proper NPSH requirements are met?

*** I'm not sure about Supp. Chamber water level at 0" is that acceptable?

A Reduce "B" Core Spray Loop flow.

B Secure "A" RHR Pump.

C Secure "A" Core Spray Loop.

D Reduce "B" RHR Pump flow.

Answer D

References
Hope Creek Question - Q53616
HC.OP-EO.ZZ-101, Reactor Pressure Vessel Control
BWR Owners Group EPGs/SAG Appendix B - Section 5 - Cautions

Justification

References during Exam None

JUSTIFICATION:
- CORRECT - Reduce "B" RHR Pump flow. "B" RHR pump is the pump for the given conditions of this question operating the closest to EOP Caution 2 NPSH limitations. The RHR flow limit is 6000 gpm.
- INCORRECT - Reduce "B" Core Spray Loop flow. No NPSH limitation concerns and RPV level is at -100", need to maintain "B" Core Spray Loop flow.
- INCORRECT - Secure "A" RHR Pump.No NPSH limitation concerns and RPV level is at -100", need to maintain "A" RHR Pump in service.
- INCORRECT - Secure "A" Core Spray Loop. "A" Core Spray Loop flow is within NPSH limits.

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:
SXD review 7/21 - OK

Question 14		Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 1	Group # 1	
☐ SRO	Importance 3.9		

295028 High Drywell Temperature / 5

EG2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Question

Given the following conditions:

·A Large Break LOCA has occurred in the Drywell concurrent with a LOP
·Only "C" EDG is running
·All control rods are fully inserted
·Drywell pressure is 25 psig and rising
·Drywell temperature is 310 F and rising
·Reactor pressure is 25 psig and steady
·Suppression Pool Level is 80 inches and rising
·"C" RHR Pump has been injecting LPCI flow for 3 minutes
·All RPV level indicators have failed upscale

Based on the above conditions, which one of the following actions is REQUIRED?

A Stop LPCI injection because adequate core cooling is assured

B Continue LPCI injection because Drywell Spray is required

C Continue LPCI injection because adequate core cooling is not assured

D Stop LPCI injection because Drywell Spray is required

Answer C *References* Hope Creek Question - Q56161, EOP- Caution 1, LP 0302-000.00H-00134-13 Obj 8
HC.OP-EO.ZZ-0206, RPV Flooding
HC.OP-EO.ZZ-0101, RPV Control

Justification *References during Exam* None

CORRECT - Continue LPCI injection because adequate core cooling is not assured. Adequate core cooling is not assured because RPV level indication are failed upscale and the criteria in RPV flooding for RPV Pressure > 50 psig above Supp Pool pressure has not been met. "C" RHR pump is not capable of injecting enough water to accurately reflect RPV water level indication off scale high.EOP-206 should be entered and LPCI injection should be continued.
INCORRECT - Stop LPCI injection because adequate core cooling is assured. Adequate core cooling is not assured because RPV level indication is in the unreliable region of EOP Caution 1. LPCI should be continued.
INCORRECT - Stop LPCI injection because Drywell Spray is required. "C" RHR pump can not be used to spray the Drywell.
INCORRECT - Continue LPCI injection because Drywell Spray is required. "C" RHR pump can not be used to spray the Drywell.

Question Source Bank ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review 7/ 21 - OK
JD 8/2 - K/A - Locate & Operate - asked to write question to J. Munro about Locate & Operate question.

Question	15	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3.5

295030 Low Suppression Pool Wtr Lvl / 5

EK3.07 Knowledge of the reasons for the following responses as they apply to Low Suppression Pool Wtr Lvl — NPSH considerations for ECCS pumps:(CFR: 41.5/41.10/45.6/ 45.13)

Question

The plant has experienced a transient and the following is observed:

- Suppression Chamber Overpressure: 9 psig
- Suppression Pool temperature: 240 degrees F
- Suppression Pool level at 74.5"
- Reactor pressure: 1000 psig
- RHR "A" pump flow: 10,000 gpm
- Core Spray "B" pump Flow: 1500 gpm
- All other low pressure ECCS pump are NOT in service.

Use the attached curves to determine if Net Positive Suction Head (NPSH) requirements are being met.

A There is sufficient NPSH for the "B" Core Spray Pump ONLY.

B There is sufficient NPSH for the "A" RHR pump ONLY.

C There is sufficient NPSH for both the "A" RHR pump and the "B" Core Spray Pump.

D There is NOT sufficient NPSH for either the "A" RHR pump or the "B" Core Spray pump.

Answer A — **References** INPO Question 14383
EOP CAUTION 2

Justification — **References during Exam** EOP Caution 2

Using EOP Caution 2 and realizing that being above the curve is the area of Unacceptable operation:
The limiting temperature for CS pump at 5 psig and 1500 gpm = 232°F
The limiting temperature for CS pump at 10 psig and 1500 gpm = 244°F
Interpolating for 9 psig gives a Temperature limit of ~242°F for 9 psig. Since given temperature = 240°F this puts the B CS pump in the area of ACCEPTABLE operation.
The limiting temperature for RHR pump at 10 psig is 235°F, since Given temperature is 240°F this puts the pump in the region of UNACCEPTABLE operation.

This makes ONLY Answer A CORRECT.

Question Source Mod — ☐ Memory Level ☑ Comprehension Level

Question History:
SXD reviewed 7/22 - OK

Question	16

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 4

295031 Reactor Low Water Level / 2

EK2.10 Knowledge of the interrelations between Reactor Low Water Level and the following — Redundant reactivity control

Question

Given the following:

-The plant is operating at 100% power.
-A transient results in a scram setpoint being exceeded.
-The Reactor Protection System fails to automatically scram the Reactor.

Without operator action, which of the following describes how the Control Rods will be automatically inserted to shutdown the Reactor via the ARI system?

A RPV level less than or equal to minus 38 (-38) inches will immediately ENERGIZE the ARI valves to depressurize the scram air header.

B RPV level less than or equal to minus 38 (-38) inches will immediately DE-ENERGIZE the ARI valves to depressurize the scram air header.

C RPV pressure greater than or equal to 1037 psig will immediately ENERGIZE the ARI valves to depressurize the scram air header.

D RPV pressure greater than or equal to 1037 psig will immediately DE-ENERGIZE the ARI valves to depressurize the scram air header.

Answer A

References INPO Question 22776
NOH01RRCS00-00, REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS), p.8

Justification

References during Exam None

A - CORRECT - with RPV level < -38" the ARI valves are energized to depressurize the scram air header resulting in rod insertion.
B. - INCORRECT - valves are Energized to actuate, not de-energized.
C. - INCORRECT - ARI pressure setpoint is 1071 psig, not 1037 psig
D - INCORRECT - ARI pressure setpoint is 1071 psig, not 1037 psig.

Question Source Mod ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review - 7/21 - Add (via the ARI system) to the end of the stem. Removed "control rod insertion will begin within 15 ...) from all distractors

Question	17	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3.8

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1

EA1.02 Ability to operate and / or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown — RRCS

Question

The plant was operating at 98% power when a transient occurred. Following the transient all SRVs opened. 2 minutes later, Reactor pressure is stable with 6 SRVs open. No operator actions have been taken.

Which of the following is correct for these conditions?

Both Recirculation Pumps _________

A have tripped.

B are running normally.

C are running at minimum speed

D are currently running but will trip in 1.9 minutes when a time delay times out.

Answer A

References INPO Question 23485
NOH01RRCS00-00, REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS), p.13

Justification

References during Exam None

A. CORRECT - Following the transient, all SRVs opened. Reactor pressure has to be greater than 1071 psig for all valves to open. Reactor pressure greater than 1071 psig causes both Recirc Pumps to trip. A is the only correct answer.
B. INCORRECT - plausible because may not have hit a trip condition.
C. INCORRECT - plausible because recirc pumps have runbacks, operator may incorrectly believe a runback condition has been met.
D - INCORRECT - plausible because a 3.9 minute timer does exist on RRCS, however it is for SLC initiation, not Recirc pump trip.

Question Source Mod ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD Review 7/21 - removed # of SRV's from stem. Removed "off" from Distractor A

Question	18	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 3.9

295038 High Off-site Release Rate / 9

EK3.02 Knowledge of the reasons for the following responses as they apply to High Off-Site Release Rate — System Isolations (CFR:41.8 to 41.10/45.3)

Question

HC.OP-EO.ZZ-0103/4, Reactor Building & Rad Release Control, step RR-5, directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building, except those systems required to assure adequate core cooling and/or shutdown the reactor.

In accordance with the EOP Bases document, HC.OP-EO.ZZ-103/4. Reactor Building & Rad Release Control, these systems are specifically exempted from isolation, because:

A systems operated for RPV control are given a higher priority than stopping a rad release.

B isolation of a EOP support system requires an upgrade of the Emergency Classification.

C they are required to support alternate reactor depressurization methods.

D additional radiological consequences from them are unlikely.

Answer A

References
INPO Question 25837
BWROG, EPGs/SAGs Appendix B, section 9 Radioactivity Release control
HC.OP-EO.ZZ-103/4. Reactor Building & Rad Release Control Bases Document - p. 13 & 14

Justification

References during Exam None

Per EOP Bases document 103/104:
The objectives of RPV Control, Primary Containment Control, and the EPG contingencies are given higher priority than the objectives of Radioactivity Release Control. Systems that must be operated to perform other steps of the EPGs are therefore not isolated in this step.
A - CORRECT matches bases document
B - INCORRECT - Not in accordance with bases document
C - INCORRECT - not in accordance with bases document
D - INCORRECT - not in accordance with bases document

Question Source Bank ☑ *Memory Level* ☐ *Comprehension Level*

Question History:

SXD review 7/22 - Minor editorial changes (added procedure)

Question	19

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	1	Group #	1

Importance 2.5

600000 Plant Fire On Site / 8

AK1.01 Knowledge of the operational implications of the following concepts as they apply to the Plant Fire On Site — Fire Classifications by type (CFR: 41.8 to 41.10 /45.3) ,

Question

A fire occurs in the Upper Cable Spreading Room (Control Equipment Mezzanine Room 5403).

The installed fire protection system automatically actuates.
The room must be entered to determine if the fire has been extinguished.
(1) What is the classification of the fire that is expected in this area?

AND

(2) What safety hazard, from the automatic system actuation, should be considered prior to operators entering the Cable Spreading Room?"

A Class C - Suffocation from oxygen depletion due to the discharge of CO_2 in the area

B Class B - Suffocation from oxygen depletion due to the discharge of halon in the area

C Class C - Suffocation from oxygen depletion due to the discharge of halon in the area

D Class B - Suffocation from oxygen depletion due to the discharge of CO_2 in the area

Answer A

References
INPO Question 24855
NOH01FIRPRO-02, FIRE PROTECTION, p.55, p. 63 and p.85

Justification

References during Exam None

A- CORRECT - Class C fire due to electrical equipment in area, Suffocation due to discharge of CO_2
B - INCORRECT - not a Class B fire and no halon in that room
C - INCORRECT - not expecting to get a halon discharge in that room
D - INCORRECT - not a class B fire

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:
SXD review - 7/21 - Changed water to Halon

Question	20	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 Group # 1

Importance 3.3

295005 Main Turbine Generator Trip / 3

AK2.04 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection (CFR: 41.7/45.8)

Question

Given the following conditions:
·The plant is operating at 20% power
·A main generator load reject has just occurred
·A fault in the control circuit causes a power/load unbalance trip during the load reject

Which of the following is the immediate expected response of the Turbine Control Valves (TCVs) and the Reactor Protection System (RPS)?

A TCVs throttle close, RPS trips

B TCVs throttle close, RPS does not trip

C TCVs fast close, RPS trips

D TCVs fast close, RPS does not trip

Answer D

References Hope Creek Question - Q61307,
HC.OP-AB.BOP-0002 Additional Information /Automatic actions and notes
NOH01MNTURB-02, MAIN TURBINE CONSTRUCTION AND COMPONENTS, p. 66

Justification

References during Exam None

CORRECT - TCVs fast close, RPS does not trip. The load reject causes the TCVs to fast close. The fast closure does not initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, no pressure transient will trip RPS.
INCORRECT - TCVs throttle close, RPS does trip. The load reject causes the TCVs to fast close. The fast closure does not initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, no pressure transient will trip RPS.
INCORRECT - TCVs fast close, RPS does trip. The fast closure does not initiate a RPS trip because turbine load is <30%. Since power is within the capacity of the BPVs, no pressure transient will trip RPS.
INCORRECT - TCVs throttle close, RPS does not trip. The load reject causes the TCVs to fast close

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:
SXD review - 7/21 - OK

Question	21

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 Group # 2

Importance 3.2

295002 Loss of Main Condenser Vac / 3

AA2.02 Ability to determine and interpret the following as they apply to Loss of Main Condenser Vacuum's Reactor Power — Plant Specific:(CFR: 41.10/43.5/ 45.13)

Question

Given the following:
·All four Circulating Water Pumps are in operation
·Plant is operating at 100% power
·Circulating Water System Inlet temperature is 80°F
·Indicated Main Condenser pressure is 2.75 in HgA

Assume the remaining Circulating Pumps' Discharge Valves are reopened fully, NO rise in basin temperature and no other operator actions are taken.

What is the expected condenser backpressure and what is the expected change in reactor power following the removal of Circulating Water Pump AP501 from service?

A 3.5 in HgA, reactor power increases (ie. greater than 2%)

B 3.5 in HgA, reactor power stays the same (ie. Doesn't change more than 2%)

C 4.15 in HgA, reactor power increases (ie. Greater than 2%)

D 4.15 in HgA, reactor power stays the same (ie. Doesn't change more than 2%)

Answer B

References Hope Creek Question - Q55132
HC.OP-SO.DA-0001, Rev. 35, Attachment 5

Justification

References during Exam Attachment 5 from HC.OP-SO.DA-0001

A - INCORRECT - Reactor power should not change with a decrease in vacuum. If anything reactor power may go down a little bit due to increased condenser temperature and reduced condenser subcooling
B- CORRECT- 3.5 inHgA. If CW inlet temp does not change, then the condenser vacuum rises vertically on the graph until it reaches the line for three pump operation @ 80 degF. Since the inital back-pressure of 2.75 indicates 100 percent CF. Reactor power should remain the same
C - INCORRECT 4.15 - 3 pump ops at 70 percent CF.
D - INCORRECT 4.15 - 3 pumps ops at 70% CF

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review 7/21 - OK
JD 8/2 - K/A asking for Reactor Power
8/3 - initially was going to change question to add reactor power change, decided to ask Steve on Monday
8/4 Re-wrote questions

Question	22	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 1 Group # 2		
	Importance 3.4		

295008 High Reactor Water Level / 2

AK3.06 Knowledge of the reasons for the following responses as they apply to High Reactor Water Level — RCIC Turbine Trip

Question

During a transient, the RO started the RCIC system for reactor water level control using the appropriate operating procedure. The RO became distracted and allowed level to rise above the High Reactor Water level at 58" after which it lowered below the Low Reactor Water level at -38".

Which of the following describes the reason for, and expected response of RCIC during the reactor water level transient?

A The RCIC Trip and Throttle Valve (HV-4282) will close on High Water Level and RCIC will automatically restart on Low Reactor Water Level.

B The RCIC Trip and Throttle Valve (HV-4282) will close on High Water Level and RCIC will have to be reset and manually started on Low Reactor Water Level.

C The RCIC Steam Supply Valve (F045) will close on High Water Level and RCIC will automatically restart on Low Reactor Water Level.

D The RCIC Steam Supply Valve (F045) will close on High Water Level and RCIC will have to be reset and manually started on Low Reactor Water Level.

Answer C **References** NOH01RCIC00-02, REACTOR CORE ISOLATION COOLING SYSTEM, p22-23

Justification **References during Exam** None

A - INCORRECT - Trip and Throttle valve does not close on Level 8
B - INCORRECT - Trip and Throttle valve does not close on Level 8
C - CORRECT - Steam supply valve will close and RCIC will auto restart at Level 2
D - INCORRECT - RCIC will auto restart at Level 2

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:

SXD review 7/22 - LOD = 1 - re-write question
8/3 - re-wrote question

Question	23	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 1 *Group #* 2

Importance 3

295009 Low Reactor Water Level / 2

AK1.02 Knowledge of the operational implications of the following concepts as they apply to the Low Reactor Water Level — Recirculation pump net positive suction head

Question

The plant is currently at 27% power. Plans for the shift are to continue the startup and power ascension. A malfunction in the Feedwater Control System has resulted in the following:

- RPV level is 25 inches and trending down
- Total Feedwater flow is 2.5 mlb/hr and steady
- 3 Circ Water pumps are running
- Condenser Vacuum is 3.8" HgA and degrading

Assume no operator actions have been taken. Which of the following statements is correct regarding the Reactor Recirculation system response based on these CURRENT plant conditions?

A Speed Limiter 1 (30% flow) is actuated to ensure Recirculation Pump net positive suction head protection based on RPV level.

B Speed Limiter 2 (45% flow) is actuated to ensure Recirculation Pump net positive suction head protection based on RPV level.

C Speed Limiter 2 (45% flow) is actuated to bring Condenser Vacuum back to normal.

D Speed Limiter 1 (30% flow) is actuated to bring Condenser Vacuum back to normal.

Answer A *References* New Question
NOH01RECIRC-02, Reactor Recirculation System, P. 53-55

Justification *References during Exam* None

A - CORRECT - Total FW flow is ~17% which is < 20%, this causes a Speed Limiter #1 runback to ensure Recirc Pump NPSH
B - INCORRECT - Speed Limiter 1 is actuated, not Speed Limiter 2
C - INCORRECT - Speed Limiter 1 is actuated, not Speed Limiter 2
D - INCORRECT - Condenser vacuum is rising but still within normal limits. Must be > 4.5" to cause a Recirc pump runback.

Question Source New ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
SXD Review 7/21 - minor editoral comments

Question	24	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 1 Group # 2	Importance 3.1	

295029 High Suppression Pool Wtr Lvl / 5

EK2.07 Knowledge of the interrelations High Suppression Pool Wtr Lvl and the following — Drywell/ containment water level:(CFR: 41.7 /45.7/45.8)

Question

An Override step in HC.OP-EO.ZZ-0202, Emergency Depressurization, directs the operator to open the Inboard MSL Drain Valve (AB-HV-F016) when Containment water level is expected to exceed 48 feet.

Which one of the following describes the reason for this action?

Opening the Inboard Main Steamline Drain Valve ______________

A maintains the availability of the Main Steamline drain path for reactor vessel pressure control if required.

B ensures as much heat energy as possible is rejected to the Main Condenser to minimize the dynamic loading on Containment.

C maintains Containment water level below the SRV solenoids by establishing a drain path from the reactor vessel to the Main Condenser.

D ensures the SRV Tail Pipe Level Limit is not exceeded prior to emergency depressurization.

Answer A

References
INPO Question 21944
BWROG EPG/SAG's App. B - P 326
HC.OP-EO.ZZ-0202 flowchart
HC.OP-EO.ZZ-0202, Emergency Depressurization Bases, p.5

Justification — **References during Exam** None

A - CORRECT - per the BWROG guidelines - If primary containment water level rises above the elevation of the SRV solenoids, the SRVs may no longer be operable. Other methods must then be used to control RPV pressure and prevent repressurization. Opening the inboard main steam line drain valve preserves the main steam line drains for future use.
B - INCORRECT but plausible, while Opening AB-HV-F016 does not reject any heat to the Main Condenser it could reject heat to the condenser if the F019 and F021 were open.
C - INCORRECT but plausible, while opening AB-HV-F016 does not necessarily maintain CNMT water level below the SRV solenoids, it may open a drain path to the main condenser.
D - INCORRECT but plausible, while opening AB-HV-F016 does not drain water from the steam lines, it could if both F019 and F021 were open.

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:
SXD reviewed 7/22 - OK

Question 27		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 1 *Group #* 2		
☐ *SRO*	*Importance* 3.1		

500000 High CTMT Hydrogen Conc. / 5

EK2.02	Knowledge of the interrelations between High CTMT Hydrogen Conc. And the following	Containment oxygen monitoring systems(CFR: 41.7 / 45.7 /45.8)

Question

Given the following conditions:

Hope Creek has experienced a transient and the following conditions are present:

Drywell H2 concentration is reading 1.5% by volume
Drywell O2 concentration is reading 5.5% by volume
Drywell Pressure is 2.0 psig and stable
Reactor water level is +10" and rising slowly (lowest level ~ 0")

Assuming no other operator actions have occurred, what is the status of the O2 monitors?

Assuming the above readings are correct and Containment venting cannot be performed, what actions shall be taken with regards to the H2 Recombiners in accordance with HC.OP-EO.ZZ-0102, Primary Containment Control?

A O2 monitors are OPERABLE and the H2 Recombiners should be placed in service.

B O2 monitors are INOPERABLE because of Containment Isolation, however the H2 Recombiners should be placed in service.

C O2 monitors are INOPERABLE because of a Containment Isolation, however the H2 Recombiners should NOT be placed in service

D O2 monitors are OPERABLE and the H2 Recombiners should NOT be placed in service.

Answer B

References
NOH01H202AN-01, Hydrogen Oxygen Analyzer System - p. 17
NOH01H2RECM-00, CONTAINMENT HYDROGEN RECOMBINER SYSTEM, p.8
HC.OP-EO.ZZ-0102(Q)-FC, PRIMARY CONTAINMENT CONTROL, step PC/H-1

Justification

References during Exam None

A. INCORRECT - O2 monitors are INOPERABLE due to a Contaiment Isolation on High Drywell Pressure.
B. CORRECT - H2 Recombiners should be placed in service due to High H2 concentration per EOP 102, concentration > 0.5% and < 2%
C. INCORRECT - H2 Recombiners should be placed in service due to High H2 Concentration per EOP 102
D. INCORRECT - O2 monitors are INOP

Question Source New ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
SXD review - 7/21 - OK

Question 29		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 2	*Group #* 1	
☐ *SRO*	*Importance* 3.5		

205000 Shutdown Cooling

A3.03 Ability to monitor automatic operations of the Shutdown Cooling System(RHR Shutdown Cooling Mode) including lights and alarms (CFR:41.7/45.5)

Question

Given the following Plant conditions:

Hope Creek is in OPCON 3 Cooling down to for a Refueling Outage,
"A" Shutdown Cooling is being placed in service and is currently in the following status:

"A" RHR fill and vent has been completed. However, the F007A - RHR Pump mini-flow valve's breaker was inadverntantly left closed.
"A" RHR Loop has been warmed up.
Both Reactor Recirc Pumps have been secured.

The RO is lining up "A" RHR system for Shutdown cooling and valves are currently lined up as follows:

F009 - Shutdown Cooling INBD ISLN MOV - Open
F008 - Shutdown Cooling OUTBD ISLN MOV - Open
AP202 RHR PUMP - Running
F015A - RHR Loop A Ret to Recirc - Throttled Open
F007A - "A" RHR pump mini-flow - Closed
F024A - "A" RHR Full Flow test valve - Closed
F027A - "A" Torus Spray Inj valve - Closed

To reduce an RCS cooldown the RO throttles closed on F015A when the following alarm is received.

"RHR A S/D CLG & MIN FL VLV OPEN" alarm is received in the control room.

Assuming NO Operator actions are taken, which of the following conditions will result:

A F008 and F009 will Auto close once the Mini-flow valve F007A gets full Open.

B F008 and F009 will Auto Close on Low RPV level 3 (+12.5")

C No Auto Actions will occur, this is an expected alarm for the above conditions.

D F008 and F009 will Auto Close on Low RPV level 1 (-129")

Answer B *References* NOH01RHRSYSC-03, RESIDUAL HEAT REMOVAL SYSTEM, p. 30

Justification *References during Exam* None

A - INCORRECT - F008 and 9 will NOT Auto Close based on mini-flow valve position
B - CORRECT - Having the Mini-flow valve open and taking suction from Reactor vessel will cause Reactor Vessel to lower, when vessel level reaches Low RPV Level 3, F008 and 009 will Auto Close.
C - INCORRECT - Reactor vessel will lower due to Mini-flow open and taking suction Reactor vessel.
D - INCORRECT - F008 and F009 will auto close on Low RPV level 3 and level should not get to Low RPV level 1.

Question Source New ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
SXD review 7/27 - OK

Question	31	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	2	Group #	1

Importance 3.3

206000 HPCI

K5.05 Knowledge of the operational implications of the following concepts as they apply to the HPCI — Turbine speed control

Question

Given the following conditions:
·The HPCI system running in automatic at rated flow.
·The flow element providing feedback to the flow controller begins to fail downscale, slowly.

How will actual HPCI turbine speed and system flow respond?

A Turbine speed will increase and flow will increase

B Turbine speed will decrease and flow will decrease

C Turbine speed will decrease and flow will remain at rated

D Turbine speed will increase and flow will remain at rated

Answer A **References** Hope Creek Question Q56448
NOH01HPCI00-02, HIGH PRESSURE COOLANT INJECTION SYSTEM, p.30

Justification **References during Exam** None

Correct answer:turbine speed will increase and flow will increase

The following distractors are incorrect as follows:
·turbine speed will increase and flow will remain at rated-Incorrect- As flow feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise
·turbine speed will decrease and flow will decrease-Incorrect- As flow feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise
·turbine speed will decrease and flow will remain at rated-Incorrect- As feedback lowers, controller will raise turbine speed and, with it actual flow rate will raise

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:
SXD review - 7/21 - OK

Question	32

Exam-Cross-Ref — Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	2	Group #	1

Importance 3

209001 LPCS

K2.01 Knowledge of electrical power supplies to the following Pump power (CFR41.7)

Question

Hope Creek has experienced a transient and a partial loss of Offsite power.

Current conditions are as follows:

Red Lion Transmission Line (61x50) is DE-ENERGIZED with a ground fault on it
500KV Circuit Breaker BS1-3(61x) failed to OPEN
13.8KV Circuit Breaker BS1-2 failed to OPEN

Reactor has SCRAMMED and all rods are INSERTED
Reactor water level is -135" and rising slowly
Drywell Pressure is 1.35# and lowering slowly (Max. Pressure ~1.5#)
"C" CS pump NORMAL/EMERGENCY TAKEOVER switch is(was) in the EMERGENCY position
A and B Diesel Generators FAILED TO START

Based on the above conditions, what is the status of the Core Spray Pumps?

A All Core Spray Pumps are running

B A, B, and D Core Spray Pumps are running

C Only C Core Spray Pump is running

D Only D Core Spray Pump is running

Answer A

References
NOH01CSSYS0-01, CORE SPRAY SYSTEM p.16
NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION
066-01: Class 1E AC Power Distribution (Training drawing)
027-01: Core Spray System (Training Drawing)

Justification

References during Exam 1.13.8KV Ring Bus – [AV1593E.vsd]

A: INCORRECT - "C" CS pump will not have started because it's Takeover switch is in the EMERGENCY Position
B: CORRECT - The Loss of the Red Lion Line and the Circuit breaker faults will have caused a loss of Bus Section 10X and Station Service XFMR 1BX501, however 1AX501 will still be energized from Offsite power, therefore power to 10A402 and 10A404 will auto transfer to 1AX501 causing all of the 4.16KV buses to be energized. As stated above "C" CS pump will not have started, leaving A, B and D CS pumps running.
C: INCORRECT - "C" CS pump will not have started because it's Takeover switch is in the EMERGENCY Position
D: INCORRECT - A and B Diesel Generators failing to start will not cause their respective buses to be de-energized because they will have received power from 1AX501

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:
SXD reviewed 7/22 - give students 500KV switchyard print

Question 33	*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO* ☐ *SRO*	*Tier #* 2 *Group #* 1 *Importance* 3.8	

211000 SLC

K4.04 Knowledge of SLC design feature(s) and or interlock(s) which provide for the following — Indication of fault in explosive valve firing circuits (CFR41.7)

Question

Hope Creek was operating at full power when an instrument air line break caused the outboard MSIVs to go closed. The following then occurred:

- The reactor failed to scram and attempts to drive rods were unsuccessful.
- The Shift Supervisor ordered SLC injection.
- Both SLC pump AP208 and BP208 START pushbuttons have been depressed.
- SLC pump control bezel start pushbuttons are backlit RED.
- The squib valve continuity lights are lit.
- Pump discharge pressure is 1395 psig.
- Reactor Pressure is currently 1025 psig.

Based on these indications which of the following correctly describes the status of the SLC system?

A SQUIB valves are closed, with SLC pumps running therefore, SLC is NOT injecting.

B SQUIB valves are OPEN, with SLC pumps running, therefore SLC is injecting

C SQUIB valves are OPEN, however, the SLC pumps are NOT running, therefore SLC is NOT injecting

D SQUIB valves are closed AND SLC pumps are NOT running, therefore, SLC is NOT injecting

Answer A

References
INPO Question 20790
NOH01SLCSYS-00, STANDBY LIQUID CONTROL SYSTEMS, p.27-29

Justification

References during Exam None

A - CORRECT - the pump control bezel start pushbuttons backlit RED, along with pump discharge pressure of 1395 psig indicate the pumps are running. Squib valve continuity lights being lit, indicate valves are closed, therefore no injection is occurring.
B - INCORRECT - Squib valves are closed
C - INCORRECT - Squib valves are closed
D - INCORRECT - SLC pumps are running

Question Source Mod ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review 7/21 - Minor editorial changes

Question	34	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 2	Group # 1	
☐ SRO	Importance 3		

212000 RPS

K3.11 Knowledge of the effect that a loss or malfunction of the RPS will have on the following — Recirculation system (CFR41.7/45.6)

Question

Given the following:

·The Reactor is initially at 20% power
·The Main Turbine is synchronized to the grid and loaded
·The RX RECIRC PUMPS RPS TRIP BYP alarm (C1-E3) is NOT illuminated
·A loss of "B" RPS Bus has occurred

What is the operational effect of a fast closure of all Turbine Control Valves during this condition?

A EOC-RPT trip of Recirculation Pump A and NO trip of Recirculation Pump B

B EOC-RPT trip of both Recirculation Pumps

C EOC-RPT trip of Recirculation Pump B and NO trip of Recirculation Pump A

D Both Recirculation Pumps running with half-scram inserted

Answer B

References Hope Creek Question - Q61263,
HC.OP-AB.ZZ.IC-0003 discussion section step 2
NOH01RECIRC-02, Reactor Recirculation System, p.37 and p. 69

Justification

References during Exam None

Justification:
·EOC-RPT trip of both Recirculation Pumps - Correct, loss of RPS bus power, at any reactor power level, in conjunction with the cited Turbine Control Valve fast closure will result in EOC-RPT trip of both Recirculation Pumps. This occurs due to a loss of the automatic bypass for EOC-RPT when less than about 30% power (first stage pressure less than 135.7 psig). The keylock bypass of the EOC-RPT trip is removed with the Main Turbine loaded. The RX RECIRC PUMPS RPS TRIP BYP alarm is cleared when the RECIRC PUMP TRIP A/B SYSTEM DISABLE switch is placed in the NORM position. This defeats the bypass of the RPT trips.
·EOC-RPT trip of Recirculation Pump A and NO trip of Recirculation Pump B - Incorrect, both pumps will trip.
·EOC-RPT trip of Recirculation Pump B and NO trip of Recirculation Pump A - Incorrect, both pumps will trip.
·Both Recirculation Pumps running with half-scram inserted - Incorrect, both pumps will trip.

Question Source Bank ☐ Memory Level ☑ Comprehension Level

Question History:
SXD Review 7/21 - OK

Question 36		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 2 *Group #* 1		
☐ *SRO*	*Importance* 2.5		

215003 IRM

K2.01 Knowledge of electrical power supplies to the following IRM Channels/ detectors (CFR41.7)

Question

A Loss of 24VDC occurs for 1AD307 DC Distribution Panel.

Which of the following describes the effect on NI's:

	SRM	IRM	APRM
A	no change	fails low	no change
B	fails low	no change	no change
C	fails low	fails low	no change
D	fails low	fails low	fails low

Answer C ***References*** NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION, p.38
NOH01IRMSYS-01, Intermediate Range Monitoring System, p26
Simplified Training prints for SRM, IRM and APRMs

Justification ***References during Exam*** None

A - INCORRECT - SRM's are powered from 24VDC and would fail downscale
B - INCORRECT - IRM's are powered from 24VDC and would fail downscale
C - CORRECT - SRM's and IRM's are powered from 24 VDC and would fail downscale, APRM's are powered from 120 VAC panels and would remain unchanged
D - INCORRECT - APRM's are powered from 120 VAC and would not fail downscale

Question Source New ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

SXD Review - 7/21 - LOD 1.0 - rewrite question to make it more difficult
8/3 - Re-wrote question

Question	41

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ **RO**
☐ **SRO**

Tier #	2	Group #	1

Importance 3.9

223002 PCIS/Nuclear Steam Supply Shutoff

A4.02 Ability to manually operate and/or monitor in the control room Manually initiate the system (CFR:41.7/45.5 to 45.8)

Question

Select the action(s) that will close all the NS4 outboard isolation valves other than the MSIVs.

A "B" and "C" NS4 logic channels are deenergized.

B "B" NS4 logic manual initiation collar is armed and pushbutton is depressed.

C "A" and "D" NS4 logic channels are deenergized.

D "D" NS4 logic manual initiation collar is armed and pushbutton is depressed.

Answer D

References
Hope Creek Question - Q53931
NOH01NSSSS0-00, NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) - p.10, p.13
Training Print 045-01: Nuclear Steam Supply Shutoff System

Justification

References during Exam None

IAW B21-1090-0062 and HC.OP-SO.SM-0001 -
A - INCORRECT - this will cause a full group one [MSIV] isolation [e.g. MSIV's will close]
B - INCORRECT - this will cause no isolation
C - INCORRECT - this will cause a full NS4 isolation and the MSIV's will close
D - CORRECT - "D" NSSSS logic manual initiation collar is armed and push-button is depressed.-Correct

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:

SXD review 7/21 - Minor editorial change - LOD 1.5 - evaluate making question more difficult

Question 42		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO*	*Tier #* 2 *Group #* 1		
☐ *SRO*	*Importance* 3.9		

239002 SRVs

A4.06 Ability to manually operate and/or monitor in the control room Reactor water level (CFR: 41.7/45.5 to 45.8)

Question

The plant is operating at 100% power, with the following:

- Reactor water level is 35 inches
- An SRV inadvertently opens

With NO operator action, which one of the following describes Reactor Water level response?

Reactor Water level will:

A lower and then return to 35 inches

B lower and remain below 35 inches

C rise and then return to 35 inches

D rise and remain above 35 inches

Answer C *References* Hope Creek Question ID - 22077
NOH01FWCONTC-02, FEEDWATER CONTROL SYSTEM, p.11

Justification *References during Exam* None

A - INCORRECT - lower and then return to 35 inches (see answer C)
B - INCORRECT - lower and remain below 35 inches (see answer C)
C - CORRECT - rise and then return to 35 inches. RPV Swells up on the RPV pressure reduction when the SRV initially opens. RPV level returns to 35 inches due to DFCS setpoint of 35 inches.
D - INCORRECT - rise and remain above 35 inches

Question Source Bank ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
SXD review 7/21 - Minor Editorial changes

Question	45	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 1	Importance 3.1	

262001 AC Electrical Distribution

K4.03 Knowledge of AC Electrical distribution design feature(s) and or interlock(s) which provide for the following — Interlocks between automatic bus transfer and breakers (CFR:41.7)

Question

With the plant in a normal electrical lineup for 100% power, the TRIP pushbutton is pressed for breaker 52-40201, Normal Feed Breaker for 10A402 on Control Room panel 10C651E.

Which choice below describes the response of the 10A402 Bus and "B" EDG?

A The Alternate Feed Breaker, 52-40208 will close energizing Bus 10A402."B" EDG will not be running.

B Bus 10A402 will be de-energized. The "B" EDG will NOT be running.

C Bus 10A402 will be de-energized. The "B" EDG will be running with its output breaker open.

D The "B" EDG will start and its output breaker will close energizing Bus 10A402.

Answer B **References** Hope Creek Question - Q53557, NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION, p.27

Justification **References during Exam** None

CORRECT - Bus 10A402 will be de-energized. The "B" EDG will NOT be running. The automatic transfer to the alternate feed and the start of the Diesel will not occur if the normal breaker is manually tripped.
INCORRECT - The Alternate Feed Breaker, 52-40208 will close energizing Bus 10A402."B" EDG Lockout will prevent the EDG start and output breaker closure. The automatic transfer to the alternate feed will not occur if the normal breaker is manually tripped.
INCORRECT - Bus 10A402 will be de-energized.The "B" EDG will be running with its output breaker open. The automatic start of the Diesel will not occur if the normal breaker is manually tripped.
INCORRECT - The "B" EDG will start and its output breaker will close energizing Bus 10A402.The automatic start of the Diesel will not occur if the normal breaker is manually opened

Question Source Bank ☑ Memory Level ☐ Comprehension Level

Question History:

SXD review 7/21 - minor editorial changes

Question	46	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 1	Importance 2.8	

262002 UPS (AC/DC)

K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the UPS (AC/DC) — DC electrical power (CFR:41.7/45.7)

Question

Hope Creek is at 100% power with the following lineup on 120V Class 1E Cyberex 20KVA Inverter 1AD481:

CB-20 - 125V DC Power Breaker Closed
CB-201 - 480V AC Normal Power Breaker Closed
CB-301 - 480V AC Backup Power Breaker Open
Auctioneered Bypass Switch is in the BYPASS 1 Position
Manual Bypass Switch is in the NORMAL Position

An Operator inadvertently opens the CB-21 (Battery Output from Auctioneered Circuit).

What effect will that have on Class 1E Instrument Distribution Panel 1AJ481?

A Class 1E Panel 1AJ481 will be de-energized due to Auctioneered Bypass Switch being in the BYPASS 1 Position.

B Class 1E Panel 1AJ481 will be energized from 480V AC Backup Power.

C Class 1E Panel 1AJ481 will be energized from 480V AC Normal Power.

D Class 1E Panel 1AJ481 will be de-energized due to CB-301 - 480V AC Backup Power Breaker being Open.

Answer C **References** NOH01EAC00-02, CLASS 1E AC POWER DISTRIBUTION, p. 60-62

Justification **References during Exam** Figures 6 and 8 of NOH01EAC00-02, AV2114D.vsd and AV2114F.vsd

A - INCORRECT - Auctioneer Bypass - Allows bypassing of one of the two Auctioneer Diodes (either diode can perform the design function) since either diode can perform the design function, bypassing diode 1 will have NO EFFECT.
B. - INCORRECT -Breaker CB-301 is given as OPEN and there are NO auto closures for this breaker.
C. - CORRECT - Power is normally supplied to 120V AC Distribution Panels from the Normal AC Power source -> Rectified to DC and then inverted back to AC. Since backup DC Power is lost, normal AC Power will still be available and the Distribution Panel will be powered as it normally is.
D. - INCORRECT - Panel 1AJ481 is not de-energized.

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:
SXD review - 7/22 - OK

Question	47	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 1		
	Importance 2.5		

263000 DC Electrical Distribution

A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical distribution controls including — Battery charging/discharging rate (CFR:41.5/45.5)

Question

"Control Room annunciator D3-F2 "125VDC SYSTEM TROUBLE" is alarming. Upon investigation the Operator determines that Digital Point D4631 "125VDC BATTERY CHARGER 1AD413" is in alarm and Battery Charger 1AD414 is INOP. On panel 10C650 the Operator reports the following:

125VDC Switchgear 10D410:
- Bus Voltage is reading 126 VDC
- Bus Current is reading 380 Amps

The following is indicated on the 125VDC Battery Charger, 1AD413, control panel:

- DC Voltmeter is reading 126 VDC
- DC Ammeter is reading 360 Amps
- Timer switch is at 0
- FLOAT light is lit
- EQUALIZING light is off
- AC PWR ON light is lit
- DC Under Voltage light if off
- DC Over Voltage light is off
- Hi Voltage Shutdown light is off
- Insufficient Charging Current light is on

WITH NO OPERATOR ACTION, which one of the following describes the expected 10D410 bus voltage trend and the reason for that trend?

The bus voltage will . . .
"

A lower because the bus load exceeds the charger's capacity.

B rise because an equalizing charge is being provided.

C rise because a malfunction of the float charge is indicated.

D lower because AC power is NOT being supplied to the charger.

Answer A

References INPO Question 24538
NOH01DCELEC-00, DC ELECTRICAL DISTRIBUTION, p25-26, p.19-20

Justification

References during Exam None

A - CORRECT - with Switchgear Load > Charger Output voltage will lower over time
B - INCORRECT - Equalizing Carge is NOT being provided with Timer switch at 0.
C - INCORRECT - Float charge is malfunctioning because charge voltage should be > bus voltage, however this will cause voltage to lower, not rise over time.
D - INCORRECT - AC on and float equalize lights indicate charger has AC power

Question Source Mod ☐ *Memory Level* ☑ *Comprehension Level*

Question History:
SXD review - 7/22 - OK

Question	49	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 2	Group # 1	
☐ SRO	Importance 2.9		

300000 Instrument Air

A3.02 Ability to monitor automatic operations of the Instrument Air including — Air temperature (CFR 41.7/45.5)

Question

Hope Creek is at 100%.

Instrument Air status is as follows:

00K107, Service Air Compressor - Disassembled for Compressor work
10K107, Service Air Compressor - Tripped due to Low Lube Oil Pressure - currently being investigated
10K100, Emergency Instrument Air Compressor - Running
Instrument Air Pressure - 90 psig stable

A SACS/TACS AUTO ISOLATION alarm is received on low pressure.

The Operators take the Mode Switch to shutdown and stablize the plant at a Reactor level of +35" (lowest level = +10").

Assuming no operator actions are taken and Instrument Air loads after the trip equal Instrument Air loads before the trip, what effect will this have on the Instrument Air system.

A It will have no effect on the Instrument Air System, instrument air pressure should be ~ equal to pre-trip value.

B Discharge air temperature will increase until the Air Compressor trips on Discharge Air Temperature high, instrument air pressure will be lower than pre-trip value.

C Cooling water supply flow will decrease until the Air Compressor trips on Low Cooling Water Supply pressure, instrument air pressure will be lower than pre-trip value.

D Reactor water level dropping to 10" causes the Air Compressor to trip on Low RPV Level, instrument air pressure will be lower than pre-trip value.

Answer A ***References*** NOH01INSAIR-01, INSTRUMENT AIR SYSTEM. P.13-14

Justification ***References during Exam*** None

A - CORRECT - Since EIAC is running and it is cooled by RACS and trips on low RPV level of -38", a loss of TACS should have no effect on EAIC and instrument air pressure should remain constant.
B - INCORRECT - EIAC is cooled by TACS, plausible distractor, if candidate thinks cooling water is isolated to compressor, discharge air temperature would increase and may cause compressor trip.
C - INCORRECT - EIAC is cooled by TACS, plausible distractor, if candidate thinks cooling water is isolated to compressor, cooling water supply flow would decrease and may cause compressor trip.
D - INCORRECT - RPV level must drop to -38" to cause EIAC to trip.

Question Source New ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:

SXD Review 7/21 - LOD - 1.0 - re-write to make more difficult
8/4 - Re-wrote question.

Question	50

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier #	2	Group #	1

Importance 2.5

300000 Instrument Air

K5.01 Knowledge of the operational implications of the following concepts as they apply to the Instrument Air — Air Compressors (CFR:41.5/ 45.7)

Question

Which statement below describes the operation of the Emergency Instrument Air Compressor if the controls are aligned for the Auto Mode.

A The compressor auto starts at 100 psig in the Emergency Instrument air receiver and loads at 100 psig and unloads at 110 psig, once auto started the compressor will run continously.

B The compressor auto starts at 70 psig in the Emergency Instrument air receiver and loads at 70 psig and unloads at 85 psig, compressor will auto stop if running unloaded for 45 minutes.

C The compressor auto starts at 85 psig in the Emergency Instrument air receiver and loads at 85 psig and unloads at 100 psig, compressor will auto stop if running unloaded for 45 minutes..

D The compressor runs CONTINUOUSLY and maintains 85-100 psig by loading and unloading.

Answer C **References** Hope Creek Question - Q54114

Justification **References during Exam** None

CORRECT: The compressor auto starts at 85 psig in the Emergency Instrument air receiver and loads at 85 psig and unloads at 100 psig.
INCORRECT: The compressor starts and loads at 70 psig and unloads at 85 psig. No: 85 to 100
INCORRECT: The compressor auto starts at 100 psig and maintains pressure between 100 and 110 psig. Wrong values. No: 85 to 100
INCORRECT: The compressor runs continuously and maintains 85-100 psig by loading and unloading. Does not run continuously. on an AUTO Start if it runs unloaded for 45 mins the compressor will STOP

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:
SXD review 7/21 - OK
MB - 7/28 - Need to add references

Question	53	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 1	Importance 3.3	

223002 PCIS/Nuclear Steam Supply Shutoff

K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/ NUCLEAR STEAM SUPPLY SHUT-OFF — Nuclear boiler instrumentation (CFR: 41.7 / 45.7)

Question

While operating RHR in shutdown cooling, reactor water level transmitter LT-N080A fails downscale.

SELECT the response of the RHR shutdown cooling supply valves, HV-F008 and HV-F009.

A Both RHR shutdown cooling supply valves will automatically close.

B Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if low level is sensed by LT-N080B.

C Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if LT-N080C fails downscale.

D Neither RHR shutdown cooling supply valve will change position automatically.

Answer D

References Hope Creek Question - Q53932
NOH01RHRSYSC-03, RESIDUAL HEAT REMOVAL SYSTEM, P.30

Justification

References during Exam None

• - Both RHR shutdown cooling supply valves will automatically close. -Incorrect - the trip must occur in both channes "a" and "b"/"c" and "d" to cause any isolation
• - Neither RHR shutdown cooling supply valve will change position automatically. Correct - the trip must occur in both channel "A" and "B" to cause an isolation
• - Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "B" NSSSS logic. -Incorrect - the trip must occur in both channels to cause any isolation. Only one would close and only when the second signal is received.
• - Only one of the RHR shutdown cooling supply valves automatically close and the second RHR shutdown cooling supply valve will close if Level 3 is sensed in the "C" NSSSS logic. -Incorrect - the trip must occur in both channels to cause any isolation

Question Source Bank ☑ Memory Level ☐ Comprehension Level

Question History:
SXD review 7/21 - OK

Question 54		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	*Tier #* 2 *Group #* 2		
	Importance 2.9		

201006 RWM

K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the RWM — Rod Position indication

Question

There is a Control Rod with an inoperable notch position reed switch. When looking at the Rod Worth Minimizer display screen for that rod, how would it's position be indicated?

A RWM would display a suggested substitute position.

B RWM would display a default value of "--"

C RWM would display the last known good position.

D RWM would display a default value of "00"

Answer A ***References*** INPO Question 1885
NOH01RODMIN-01, ROD WORTH MINIMIZER p.15

Justification ***References during Exam*** None

A - CORRECT - per Lesson Plan - If a control rod is moved to a position with a failed reed switch, the RWM program will:a)Allow a single notch insert or withdraw permissive to allow the control rod to be moved to verify its actual position. b)Suggest to the operator a substitute position, which is its calculated inferred position.
B - INCORRECT - See "A"
C - INCORRECT - See "A"
D - INCORRECT - See "A"

Question Source Mod ☑ ***Memory Level*** ☐ ***Comprehension Level***

Question History:

SXD review - 7/22 - Had questions talk to Archie about what would be displayed. Perhaps change inop notch position to a given position (ie. 12). If you pull rod from 10 to 12 and position 12's reed switch is INOP is 12 displayed.

Question	56	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO ☐ SRO	Tier # 2 Group # 2		
	Importance 3.8		

219000 RHR/LPCI: Torus/Pool Cooling Mode

K4.03 Knowledge of RHR/LPCI Torus/Pool Cooling Mode design feature(s) and or interlocks which provide for the following — Unintentional reduction in vessel injection flow during accident conditions

Question

You have the following plant conditions:

o Drywell pressure 3.2 psig
o Drywell temperature 170°F
o Suppression Pool pressure 1.8 psig
o Suppression Pool temperature 96°F
o Reactor water level + 25 inches

The plant has scrammed on high Drywell pressure and the actions of both Primary Containment Control and RPV Control are being carried out.

The RHR system was in a normal lineup at the beginning of the transient and all automatic actions occurred as designed.

The CRS orders Suppression Pooling Cooling started on the ""A""RHR Loop. Which of the following switch manipulations will have to be performed in order to start Suppression Pool Cooling On the ""A"" RHR Loop IAW HC.OP-SO.BC-0001, RHR System Operation?

A AUTO OP OVRD must be pressed on BC-HV-F017A, RHR LOOP A LPCI INJ MOV before valve can be closed. Once valve is closed then BC-HV-F024A, RHR LOOP A TEST RET MOV can be opened by depressing it's OPEN pushbutton.

B BC-HV-F017A, RHR LOOP A LPCI INJ MOV must be closed by depressing it's closed pushbutton. Once F017A is closed then BC-HV-F024A, RHR LOOP A TEST RET MOV can be opened by depressing it's OPEN pushbutton.

C AUTO OP OVRD must be pressed for BC-HV-F017A, RHR LOOP A LPCI INJ MOV prior to depressing it's CLOSED pushbutton. Once F017A is closed then AUTO CL OVRD must be pressed for BC-HV-F024A, RHR LOOP A TEST RET MOV prior to depressing it's OPEN pushbutton.

D AUTO CL OVRD must be pressed on BC-HV-F017A, RHR LOOP A LPCI INJ MOV before valve can be closed. Once valve is closed then AUTO OP OVRD must be pressed on BC-HV-F024A, RHR LOOP A TEST RET MOV prior to opening F024A.

Answer C

References INPO Question 2069
HC.OP-SO.BC-0001(Q) - Rev. 40, RESIDUAL HEAT REMOVAL SYSTEM OPERATION, p. 23, Note 5.5.5

Justification — **References during Exam** None

A - INCORRECT - AUTO CL OVRD must be pressed on F024A before valve can be opened with LPCI initiation signal present.
B. INCORRECT - must depress AUTO OP OVRD for F017A prior to closing F017A with LPCI signal present
C. CORRECT - per Procedure Note 5.5.5 - If a LPCI Initiation signal is present, the AUTO OP OVRD must be pressed on BC-HV-F017A(B) RHR LOOP A(B,C,D) LPCI INJ MOV, before the valve can be closed. The AUTO CL OVRD must be pressed on BC-HV-F024A(B) RHR LOOP A(B) TEST RET MOV, and BC-HV-F017A(B) must be closed before BC-HV-F024A(B) can be opened.
D. INCORRECT - Must Depress AUTO OP OVRD on F017A not AUTO CL OVRD

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

SXD Review 7/22 - verify pushbutton labels are correct

Question	57		Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 2	Group # 2		
☐ SRO	Importance 4.2			

239001 Main and Reheat Steam

A3.01 Ability to monitor automatic operations of the Main and Reheat system including Isolation of main steam system (CFR:41.7/45.5)

Question

The plant is shutting down for a refueling outage.

Current plant conditions are as follows:
Mode Switch - STARTUP
Reactor Power - 4%
Reactor Pressure - 1000 psig
Reactor Level - 35"
"A" RFP running
Both Recirc pumps running
Condenser vacuum - 3.5" abs
3 Circ Water Pumps running
All MSIV's open

An event occurs:
3 Minutes later plant conditions are as follows:
Mode Switch - SHUTDOWN
Reactor Power - All Rods inserted
Reactor pressure - 700 psig decreasing
Reactor Level - (-50" lowering)
Condenser Vacuum - 23" abs Degrading

Based on the above conditions and assuming no operator actions, what is the status of the MSIV's and explain the reason for that status.

A MSIV's all OPEN - No automatic closure signal exists

B MSIV's all CLOSED - due to 1 Automatic Closure signal - Low Reactor Pressure

C MSIV's all CLOSED - due to 1 Automatic Closure signal - Low Condenser Vacuum

D MSIV's all CLOSED - due to 2 Automatic Closure signals - Low Reactor Pressure and Low Condenser Vacuum

Answer C **References** NOH01MSTEAMC-02, MAIN STEAM SYSTEM p.24

Justification **References during Exam** Figure of NSSS

A - INCORRECT - Condenser Vacuum of > 21.5" will cause MSIV's to close. Plausible distractor - this isolation can be bypass with a keylock switch.
B - INCORRECT - Low Reactor Pressure MSIV closure signal is bypassed when Mode Switch is NOT in RUN
C - CORRECT - Low Condenser vacuum setpoint of 21.5" has been reached and limit has not been bypassed.
D - INCORRECT - Low Reactor Pressure MSIV closure signal is bypassed when Mode Switch is NOT in RUN

Question Source New ☐ Memory Level ☑ Comprehension Level

Question History:

SXD Review 7/21 - LOD 1.0 re-write question
8/4 - Wrote new question

Question 60		*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005
☑ *RO* ☐ *SRO*	*Tier #* 2 *Group #* 2		
	Importance 3.2		

272000 Radiation Monitoring

K5.01 Knowledge of the operational implications of the following concepts as they apply to the Radiation Monitoring — Hydrogen injection operation's effect on process radiation indications

Question

"The plant was operating at full power with indicated H2 injection flow at 10 SCFM, when FE-601 (Flow Input to Hydrogen Flow Controller - FIC-601) fails LOW (ie. A LOW flow is INPUT into FIC-601).

Which of the following describes the expected result?

The Hydrogen Flow Control Valves will..."

A open rapidly resulting in rising Main Steam Line radiation levels.

B fail as is with no adverse consequences.

C open rapidly resulting in a "LOW Recirc dissolved Oxygen level" alarm.

D close rapidly resulting in "HIGH Recirc dissolved Oxygen level" alarm.

Answer A

References
INPO Question 8753
NOH01HWC100-01, HYDROGEN WATER CHEMISTRY INJECTION SYSTEM, p. 12
M-101-0 sht 1 & 2

Justification

References during Exam None

A - CORRECT - FIC-601 attempts to maintain a certain H2 flow to the Secondary Condensate Pumps, when this flow input fails LOW - FIC-601 will attempt to raise H2 flow by opening the H2 Flow Control valves, opening these valves will result in Rising Main Steam Line Radiation Levels.
B - INCORRECT - H2 FCV's will open
C - INCORRECT - While the FCV's will open rapidly, there is NO Low Recirc Dissolved Oxygen Level alarm.
D - INCORRECT - FCV's will open.

Question Source Mod ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review - 7/27 - Minor editorial changes

Question	68

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

☑ RO
☐ SRO

Tier # 3 *Group #*

Importance 3.4

2.1.33 Generic

Ability to recognize indications for system operating parameters which are entry level condition for Technical Specifications (CFR: 43.2 / 43.3 /45.3)

Question

During Plant startup the following conditions are observed:

TIME	RPV Pressure
0700	172 psig
0715	191 psig
0730	211 psig
0745	233 psig
0800	373 psig

Which one of the following is the latest time at which heatup must be secured in order to prevent exceeding the Technical Specification limit for heatup at the CURRENT heat up rate?

A 0800

B 0815

C 0830

D 0845

Answer B

References Hope Creek Question - Q56983
Steam Tables
Tech Spec

Justification

References during Exam Steam Tables

Justification
172 psig = 186.7psia=376F
191psig=205.7 psia = 384°F
211 psig – 225.7 psia = 392F
233 psig = 247.7 psia = 400F
373 psig = 387.7 psia = 442F-This gives a 42F change in 15 mins. Current heatup rate is 42F every 15 min (168 degrees/hr). 0815 - Correct- At this rate we must terminate the H/U by 0815 to keep from exceeding the allowable heatup, we would be at 484°F (this would be 100 degrees/hr).

Question Source Bank ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review 7/21 - OK

Question	69	Exam-Cross-Ref	Hope Creek RO Exam - Nov 2005
☑ RO	Tier # 3	Group #	
☐ SRO	Importance 3.7		

2.2.1 Generic

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Question

The plant is shutdown with 'B' RHR in shutdown cooling, OPCON 4. Inservice stroke time testing needs to be performed on the discharge valve of the 'A' recirculation pump prior to commencing startup.

What precautions/limitiations exist to allow/prevent this evolution to take place?

A As long as RPV vessel level is pegged high on all Narrow Range instruments, Shutdown cooling may be secured and the recirculation discharge valve stroked without potential problem of loss of decay heat removal and vessel stratification.

B System Operating procedures for both Recirculation system and RHR system prohibit the opening of Recirculation pump discharge valves while RHR is in Shutdown Cooling, to prevent potential core bypass flow and vessel stratification.

C This evolution can only be performed after the 'B' Recirc pump is placed in service and establishment of forced circulation through the vessel is assured.

D Prior to stroking the discharge valve on 'A' Recirculation pump, the suction valve must be verified closed, and the suction valve's power supply breaker open.

Answer D **References** Hope Creek Question - Q56375
HC.OP-IO.ZZ-0002 section 3.2.5

Justification **References during Exam** None

Justification:IAW HC.OP-IO.ZZ-0002 section 3.2.5
·"Prior to stroking the discharge valve on 'A' Recirculation pump, the suction valve must be verified closed, and the supply breaker opened."-Correct
·"This evolution can only be performed after the 'B' Recirc pump is placed in service and establishment of forced circulation through the vessel is assured."- Incorrect- The 'can only' distractor is wrong because the word "only " is used, along with the combination of RHR and Recirc pump combinations would still require the suction valve closed while stoking the valve
·"System Operating procedures for both Recirculation system and RHR system prohibit the opening of Recirculation pump discharge valves while RHR is in Shutdown Cooling, to prevent potential core bypass flow and vessel stratification." - Incorrect- The 'SOP' distractor is wrong because the IO allows this condition and applicable exception to the SO guidance
·"As long as RPV vessel level is pegged high on all Narrow Range instruments, Shutdown cooling may be secured and the recirculation discharge valve stroked without potential problem of loss of decay heat removal and vessel stratification." – Incorrect- The 'RPV vessel level' is wrong because minimum level for natural circulation is +80" which is well above the Narrow Range detector capability to read, and does not assure the appropriate level.

Question Source Bank ☑ **Memory Level** ☐ **Comprehension Level**

Question History:
SXD review 7/21 - OK

Question	71	*Exam-Cross-Ref*	Hope Creek RO Exam - Nov 2005

☑ *RO*
☐ *SRO*

Tier # 3 *Group #*

Importance 2.6

2.3.1 Generic

Knowledge of 10 CFR 20 and related facility radiation control requirements (CFR: 41.12 / 43.4. 45.9 / 45.10).

Question

Radiation Protection technicians have surveyed the Refuel Floor Reactor Head Laydown Area during an outage and obtained the following results:

·Highest Area Dose Rate one foot from any source in the room: 72 mr/hr
·Airborne Concentration: 0.15 DAC
·Smear Results: 750 dpm/100 cm2 gamma

Based on these results the area should be posted as a:

I. Radiation Area
II. High Radiation Area
III. Very High Radiation Area
IV. Contaminated Area
V. Airborne Radioactivity Area

A I, and V

B I, IV, and V

C I and IV

D II and IV

Answer A *References* Hope Creek Question - Q76884 (Modified slightly)
NC.NA-AP.ZZ-0024, rev 13, p.23

Justification *References during Exam* None

A - CORRECT - Airborne rad area > 10% or .10 DAC
B - INCORRECT - Not a Contaminated Area - must be > 1000 dpm/10cm2
C - INCORRECT - Not a Contaminated area and it is an Airborne Area
D - INCORRECT - Not a High Radiation Area - must be > 100mr/hr

Question Source Bank ☑ *Memory Level* ☐ *Comprehension Level*

Question History:

SXD review 7/21 - LOD 1.5 - evaluate writing a more difficult question
Changed question out with another HC bank question that seems more difficult

Question	75	Exam-Cross-Ref	Hope Creek SRO Exam - Nov 200

☐ RO

☑ SRO

Tier # 3 Group #

Importance 3.3

2.4.31 Generic

Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)

Question

The plant is in Mode 5 with the Spent Fuel Pool/ Cavity Gates installed. Preparations are underway to flood the reactor cavity when the following annunciators are received:

D1A5 - FUEL POOL LEVEL HI/LO
D1D5 - FUEL POOL COOLING SYS TROUBLE

The Operator reports that Spent Fuel Pool level and Skimmer Surge tank level are ~21.5' and lowering. Radiation levels in the Spent Fuel pool are rising.

Which procedure should be entered and what is the preferred method to keep the fuel bundles covered?

*** Hope Creek - is it acceptable to directly enter AB or would you prefer I state per the ARP.

A HC.OP-AB.COOL-0004, FUEL POOL COOLING should be entered and Spent Fuel Pool level should be raised using the Fuel Pool Cooling System.

B HC.OP-AB.COOL-0004, FUEL POOL COOLING should be entered and Spent Fuel Pool level should be raised using the RHR system.

C HC.OP-EO.ZZ-0103/4, REACTOR BUILDING AND RADIOACTIVE RELEASE CONTROL, should be entered and Spent Fuel pool Level should be raised using the Fuel Pool Cooling System.

D HC.OP-EO.ZZ-0103/4, REACTOR BUILDING AND RADIOACTIVE RELEASE CONTROL, should be entered and Spent Fuel pool Level should be raised using the RHR System.

Answer B ***References*** HC.OP-AR.ZZ-0013, Overhead Annunciator Window Box D1, p. 36 and 89
NOH01FPCC00-03, Fuel Pool Cooling and Cleanup System, p.17

Justification ***References during Exam*** None

A - INCORRECT - Cannot use Fuel Pool Cooling for makeup because Fuel Pool Cooling pumps trip on Low skimmer surge tank level of 22'.
B - CORRECT - Based on the Annunciator Response Procedure for Low Fuel Level - AB.COOL-0004 should be entered and level raised using the RHR system because the Fuel Pool Cooling System is not available.
C - INCORRECT - even though radiation levels are rising, based on the Annunciator response procedure AB.COOL-0004 should be entered.
D. INCORRECT - even though radiation levels are rising, based on the Annunciator response procedure AB.COOL-0004 should be entered.

Question Source Mod ☐ ***Memory Level*** ☑ ***Comprehension Level***

Question History:

SXD review 7/27 - Not SRO level - re-write
Re-wrote question - 7/29 - somewhat based on INPO Question 22362
JD - 8/2 - Why are C,D plausible
MB - 8/3 - Changed AB.CONT-0005, Irradiated Fuel damage to EO.ZZ-0103/4 since Radiation levels in the Reactor Building are rising and operator may be concerned about reactor building release.

Question	76	*Exam-Cross-Ref*
☐ *RO*	*Tier #* 1	*Group #* 1
☑ *SRO*	*Importance*	3.8

295003 Partial or Complete Loss of AC / 6

AG2.1.32 Ability to explain and apply system limits and precautions
(CFR 41.10/ 43.2/ 45.12)

Question

Hope Creek was operating at 30% power when a Station Blackout (loss of all onsite and offsite power) occurred causing a Reactor Scram.

Current plant conditions are as follows:

Drywell temperature - 300°F decreasing slowly
RPV pressure - 273 psig decreasing slowly
Reactor Power - all rods fully inserted
Reactor level - (-100" decreasing)
RCIC - tagged out and disassembled
HPCI - tripped on overspeed and will not restart
"A" EDG - tagged out for maintenance
"B" EDG - running unloaded - output breaker failed open on anti-pump circuitry
"C" EDG - tripped on Bus differential overcurrent
"D" EDG - failure to start - low air pressure ~20 psig

Based on these conditions, the Control Room Supervisor shall:

A direct the NEO to reset the Bus differential overcurrent on the "C" EDG and restart the "C" EDG.

B direct the RO to depress the TRIP pushbutton on the "B" EDG output breaker and verify output breaker closes

C enter procedure HC.OP-EO.ZZ-0202, Emergency Depressurization based on high Drywell temperature.

D enter procedure HC.OP-EO.ZZ-0202, Emergency Depressurization before Reactor Water Level decreases to -129"

Answer C *References* HC.OP-AB.ZZ-0135, Station Blackout// Loss of Offsite Power// Diesel Generator Malfunction p. 2

Justification *References during Exam* None

A - INCORRECT - bus differential current should not be reset without electrical maintenance determining and correcting the cause.
B - CORRECT - per HC.OP.AB.ZZ-0135, Station Blackout p. 18 step 5.16 - The Anti-pump circuitry on the D/G output breaker could cause the output breaker to fail open. To load the D/G under this condition the operator must depress the TRIP push-button (even though the breaker is already tripped) to reset the logic. When the TRIP push-button is released, then the breaker will close and the D/G will load.
C - INCORRECT - Emergency Depressurization procedure should not be entered until DW temperature exceeds 340°F and current drywell temperature is decreasing.
D - INCORRECT - Emergency Depressurization procedure should not be entered until is less than -129" but before level decreases to -185"

Question Source New ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review 7/27 - Not SRO level - re-write
8/2 - re-wrote question

Question	80	Exam-Cross-Ref	Hope Creek SRO Exam - Nov 200
☐ RO ☑ SRO	Tier # 1 Group # 1 Importance 3.3		

295028 High Drywell Temperature / 5

EG2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR 45.3)

Question

Given the following conditions:

·A small steam leak has occured in the drywell causing a reactor scram	
·Two control rods are at position	06
·RPV level	+30 inches
·RPV pressure	920 psig
·Suppression pool level	75 inches
·Suppression pool temperature	80 °F
·Drywell pressure	3 psig
·Average drywell temperature	330 °F and rising at 1°F per minute
·Suppression chamber pressure	3 psig

Which of the following describes the next operator action(s) in accordance with the Emergency Operating Procedures?

A Shutdown the Reactor Recirculation Pumps and Drywell Cooling Fans and initiate one loop of drywell spray.

B Verify all injection into the RPV except SLC, CRD and RCIC is terminated and prevented and then emergency depressurize the reactor.

C Rapidly depressurize the reactor using the main turbine bypass valves.

D Initiate suppression chamber sprays and commence a normal reactor cooldown. (Less than 90 F per hour)

Answer B **References** Hope Creek Question Q56045
HC.OP-EO.ZZ-0102 Bases, step DW/T-5

Justification **References during Exam** None

A - INCORRECT - ·Shutdown the Reactor Recirculation Pumps and Drywell Cooling Fans and initiate one loop of drywell spray.-incorrect- Cannot DW Spray since outside of DWT-P curve.
B - CORRECT - ·Verify all injection into the RPV except SLC, CRD and RCIC is terminated and prevented and then emergency depressurize the reactor.-correct- EOP-0202 step ED-3
C - INCORRECT - ·Rapidly depressurize the reactor using the main turbine bypass valves.-incorrect- EOP-101A prevents use of BPVs in this situation
D - INCORRECT - ·Initiate suppression chamber sprays and commence a normal reactor cooldown. (Less than 90 F per hour)-incorrect- must stabilize pressure until S/D under all conditions without Boron

Question Source Bank ☐ **Memory Level** ☑ **Comprehension Level**

Question History:
SXD review - 7/29 - OK

Question	83	*Exam-Cross-Ref*	Hope Creek SRO Exam - Nov 200

☐ *RO*

☑ *SRO*

Tier #	1	*Group #*	2

Importance 3.7

295009 Low Reactor Water Level / 2

AA2.02 Ability to determine and interpret the following as they apply to Low Reactor Water Level (CFR: 41.10/ 43.5 / 45.13) Steam flow/ feed flow mismatch

Question

Hope Creek is operating at 75% power reducing power to remove the 6A Feedwater Heater from service due to a problem on the Bleeder trip valve with the following conditions:

Feedwater control is in 3 element control
A Steam Flow indicates - 2.0 E6 lbs/hr
B Steam Flow indicates - 2.0 E6 lbs/hr
C Steam Flow indicates - 2.0 E6 lbs/hr
D Steam Flow indicates - 2.0 E6 lbs/hr
FW flow (N001A) indicates - 4.0 E6 lbs/hr
FW flow (N001B) indicates - 4.0 E6 lbs/hr
Reactor Water level - 33" stable
Reactor Pressure - 1000 psig stable
Generator MW - 750 MW

An event occurs.
1 Minute after event initiation the following conditions are observed:

A Steam flow indicates - 1.7 E6 lbs/hr
B Steam flow indicates - 2.0 E6 lbs/hr
C Steam flow indicates - 2.0 E6 lbs/hr
D Steam flow indicates - 2.0 E6 lbs/hr
FW flow (N001A) indicates - 3.9 E6 lbs/hr
FW flow (N001B) indicates - 3.9 E6 lbs/hr
Reactor Water level - 38" and lowering slowly
Reactor Pressure - 990 psig stable
Generator MW - 710 MW

Based on the above conditions, what event has happened and what procedure shall you direct the operators to respond to the event?

** Hope Creek to supply valid numbers (Steam Flow/Feed flow or I'll do it when I go back down there.

A "A" Steam line's input to Total Steam flow has partially failed causing Steam flow/Feed flow mismatch, go to procedure HC.OP-AR.ZZ-0007 window F-1, "DFCS ALARM/TRBL"

B "A" Main Stop Valve has failed closed, go to procedure HC.OP-AB.BOP-0002, MAIN TURBINE

C A Safety has opened on the "A" steam line, go to procedure HC.OP-AB.RPV-0006,SAFETY RELIEF VALVE

D The 6A Feedwater heater bleeder trip valve has failed, go to procedure HC.OP-AB.ZZ-0001, TRANSIENT PLANT CONDITIONS

Answer C *References* HC.OP-AB.RPV-0006, Safety Relief Valve p.1
NOH01MSTEAMC-02, MAIN STEAM SYSTEM,

Justification *References during Exam* None

A - INCORRECT - While "A" steam line's input to Total Steam flow could cause the difference in indicated Steam Flow, it would not cause Generator MW to decrease.
B - INCORRECT - While "A" Main stop valve failing closed would cause a decrease in MW, it would not cause Reactor pressure to decrease, it would increase.
C - CORRECT - A safety on "A" steam line would cause, "A"'s steam line flow to decrease, MW to decrease and Reactor Pressure to decrease.
D - INCORRECT - 6A's bleeder trip valve going closed would cause MW to go up not down.

Question Source New ☐ *Memory Level* ☑ *Comprehension Level*

Question History:

SXD review 7/27 - LOD 1 - re-write
8/1 - re-wrote question - MB

Question	95	Exam-Cross-Ref	Hope Creek SRO Exam - Nov 200

☐ RO
☑ SRO

Tier #	3	Group #	

Importance 2.9

2.1.34 Generic

Ability to maintain primary and secondary plant chemistry within allowable limits (CFR: 41.10 / 43.5 / 45.12)

Question

The plant was operating at 20% power. Plant Chemistry reported to the Main Control Room the following chemistry parameters:

- Reactor pH	8.8
- Reactor Water conductivity	11 micromhos/cm
- Reactor Water chlorides	150 ppb

Six hours later with the plant in OPCON 2, Chemistry reports the following:

- Reactor pH	6.5
- Reactor Water conductivity	0.9 micromhos/cm
- Reactor Water chlorides	150 ppb

Which one of the following actions is appropriate for these plant conditions?

A Be in OPCON 3 within 6 hours and OPCON 4 within 30 hours.

B Return to OPCON 1 where chemistry would be back in spec.

C Stay in OPCON 2 and restore chlorides to within limits within 48 hours or be in OPCON 3 within the next 12 hours and OPCON 4 within the following 24 hours.

D Restore Chlorides to within spec within 18 hours or perform an engineering evaluation.

Answer A

References INPO Question 24577
UFSAR 5.2.3.2.2.2 and UFSAR Table 5.2-8

Justification

References during Exam UFSAR 5.2.3.2.2.2 and Table 5.2-8

A - CORRECT - per ACTION a. - with conductivity exceeding 10mmho/cm be in HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the next 24 hours.
B - INCORRECT - plausible because based on given conditions for OPCON 2, plant chemistry would be in spec if plant returned to OPCON 1.
C - INCORRECT - plausible if only look at Action b.
D - INCORRECT - plausible if only look at Action c.2

Question Source Mod ☐ Memory Level ☑ Comprehension Level

Question History:

SXD review 7/27 - Talk to licensee ensure correct answer is correct and once conductivity is < limit, they exit the condition and can return to power.

Question 98	Exam-Cross-Ref	Hope Creek SRO Exam - Nov 200

☐ RO
☑ SRO

Tier # 3 Group #

Importance 3.1

2.3.4 Generic

Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)

Question

Lowest authorization necessary to receive increase a worker's dose control level to 3500mrem/yr per NC.NA-AP.ZZ-0024, Radiation Protection Program, is the responsibility of the ____________ .

A Radiation Protection Supervisor

B Radiation Protection Manager

C Plant Manager

D Emergency Director

Answer B

References INPO Question 19298
NC.NA-AP.ZZ-0024, RADIATION PROTECTION PROGRAM - p. 27

Justification

References during Exam None

A - INCORRECT - Radiation Protection Supervisor may only increase dose level to 3000 mrem/yr
B - CORRECT - Radiation Protection Manager may raise dose control level to 4000 mrem/yr
C - INCORRECT - While Plant Manager may raise level to 4750 mrem/yr, he is not the Lowest authorization necessary.
D. - INCORRECT - While ED may approve Emergency Doses, he is also not the Lowest authorization necessary.

Question Source Mod ☑ Memory Level ☐ Comprehension Level

Question History:

SXD review 7/27 - too easy - LOD 1

Question	99

Exam-Cross-Ref Hope Creek SRO Exam - Nov 200

☐ RO
☑ SRO

Tier #	3	Group #	

Importance 4

2.4.22 Generic

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 43.5 / 45.12)

Question

During an ATWS, automatic initiation of the Automatic Depressurization System (ADS) is inhibited to prevent which one of the following?

A A power excursion due to low pressure ECCS injection

B Large irregular neutron flux oscillations

C Exceeding 110°F Suppression Pool Temperature before boron injection

D Causing a Pressurized Thermal Shock to the Reactor Vessel

Answer A *References* INPO Question 24595
HC.OP-EO.ZZ-0101A, ATWS – RPV CONTROL, P. 18

Justification *References during Exam* None

A - CORRECT - Per EOP 101A bases - Further, rapid and uncontrolled injection of large amounts of relatively cold, unborated water from low pressure injection systems may occur as RPV pressure decreases to and below the shutoff heads of these pumps. Such an occurrence would quickly dilute in-core boron concentration and reduce reactor coolant temperature. When the reactor is not shutdown, or when the shutdown margin is small, sufficient positive reactivity might be added in this way to cause a reactor power excursion large enough to severely damage the core.
B - INCORRECT - ADS initiation would NOT cause flux oscillation but rather a rapid reduction in core power due to voids
C - INCORRECT - This may or may NOT be true but it is NOT the reason for inhibiting ADS
D - INCORRECT - While an ADS actuation will cause a Thermal Shock to the vessel, the vessel will be de-pressurized so you will not have a PTS concern

Question Source Mod ☑ *Memory Level* ☐ *Comprehension Level*

Question History:
SXD review 7/27 - OK

Question	12

☐ RO

☐ SRO

Exam-Cross-Ref Hope Creek RO Exam - Nov 2005

Tier #	1	Group #	1

Importance 3.8

295025 High Reactor Pressure / 3

EA1.02 Ability to operate and / or monitor the following as they apply to High Reactor Pressure — Reactor/Turbine pressure regulating system :(CFR: 41.7/45.5/ 45.6)

Question

Given the following conditions:

·The plant is operating at power 91% power.
·The steam pressure input signal to the "A" EHC regulator fails downscale.
·No operator actions are taken.

Which of the following is the response of Reactor pressure to the conditions above?

*** NEED DIFFERENT QUESTION DUE TO DIGITAL EHC ***

A Pressure rises 3 psig and stabilizes.

B Pressure lowers to the MSIV isolation setpoint.

C Pressure lowers 3 psig and stabilizes.

D Pressure rises to the scram setpoint.

Answer A **References** Hope Creek Question - Q61768, HC.OP-AB.RPV-0005 Automatic Actions

Justification **References during Exam** None

CORRECT - Pressure rises 3 psig and stabilizes. This failure causes the TCVs to throttle closed, raising RPV pressure. When the actual pressure increase overcomes the 3 psi bias on the "B" pressure regulator, the "B" regulator will re-open the TCVs and maintain pressure 3 psig higher than the "A" regulator.
INCORRECT - Pressure lowers 3 psig and stabilizes. This failure causes the TCVs to throttle closed, raising RPV pressure.
INCORRECT - Pressure lowers to the MSIV isolation setpoint. This failure causes the TCVs to throttle closed, raising RPV pressure.
INCORRECT - Pressure rises to the scram setpoint. When the actual pressure increase overcomes the 3 psi bias on the "B" pressure regulator, the "B" regulator will re-open the TCVs and maintain pressure 3 psig higher than the "A" regulator.

Question Source Bank ☐ **Memory Level** ☐ **Comprehension Level**

Question History:

Facility: Hope Creek Date of Examination: 11/28/05

Item	Task Description	Initials a	Initials b*	Initials c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model, in accordance with ES-401.	MB	SA	SD
	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	SA	SD
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	MB	SA	SD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MB	SA	SD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.	MB	SA	SD
	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, and ensure that 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 that scenarios will not be repeated on subsequent days.	MB	SA	SD
	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. (Later)	MB	SA	SD
3. W/T	a. Verify that the systems walk-through outline meets the criteria specified on Form ES-301-2: (1) the outline(s) contain(s) the required number of control room and in-plant tasks distributed among the safety functions as specified on the form (2) task repetition from the last two NRC examinations is within the limits specified on the form (3) no tasks are duplicated from the applicants' audit test(s) (4) the number of new or modified tasks meets or exceeds the minimums specified on the form (5) the number of alternate path, low-power, emergency, and RCA tasks meet the criteria on the form.	MB	SA	SD
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) the tasks are distributed among the topics as specified on the form (2) at least one task is new or significantly modified (3) no more than one task is repeated from the last two NRC licensing examinations	MB	SA	SD
	c. 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	SA	SD
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam sections.	MB	SA	SD
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.	MB	SA	SD
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	MB	SA	SD
	d. Check for duplication and overlap among exam sections.	MB	SA	SD
	e. Check the entire exam for balance of coverage.	MB	SA	SD
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	MB	SA	SD

*	Printed Name/Signature	Date
a. Author	Michael L. Brown / Michael L. Brown	9/2/05
b. Facility Reviewer (*)		
c. NRC Chief Examiner (#)	Steven Dennis / [signature]	9/6/05
d. NRC Supervisor	Rich Conte / [signature]	9/6/05

Note: # Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

* OUTLINE ONLY

Facility: Hope Creek		Date of Exam: 11/28/05						Operating Test No.:							
APPLICANT	EVENT TYPE	Scenarios												TOTAL	MINIMUM
		1			2			3			4				
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION				
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		
RO	RX		1											1	1*
	NOR						1							1	1*
	I/C		2,3,4				1,4,5, 9							7	4*
	MAJ		5				6,8							3	2
	TS														2
SRO-I	RX	1				2								2	1*
	NOR													0	1*
	I/C	2,3,4, 5,6				3,6, 7								8	4*
	MAJ	5				6,8								3	2
	TS	3,4												2	2
RO	RX		1											1	1*
	NOR						1							1	1*
	I/C		2,3 , 4				1,4,5, 9							7	4*
	MAJ		5				6,8							3	2
	TS														2
SRO-I	RX	1				2								2	1*
	NOR													0	1*
	I/C	2,3,4, 5,6				3,6, 7								8	4*
	MAJ	5				6,8								3	2
	TS	3,4												2	2

Instructions:

1. Circle the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls (ATC)" and "balance-of-plant (BOP)" positions; Instant SROs must do one scenario, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position.

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 1-for-1 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: ______________________

NRC Reviewer: ______________________

Facility: Hope Creek Scenario No.: 1 Op-Test No.: ______

Examiners: ______ Operators: ______

Initial Conditions: 4% power, Reactor Startup in progress, "B" EHC pump blocked for maintenance

Turnover:

A Reactor Startup is in progress with IOP-3 completed up to step 5.3.21. Reactor power is approximately 4%. RCIC is being operated for HC.OP-ST.BD-0002, RCIC Pump Valve and flow test and should be completed within the next hour. 1BP116 EHC pump is tagged out for maintenance and will be out of service until a new pressure compensator arrives tomorrow.

Event No.	Malf. No.	Event Type*	Event Description
1		R (RO) R (CRS)	Withdraw control rods until 4 bypass valves open
2		I (RO) I (CRS)	CRD Flow controller fails downscale in AUTO
3		I (ALL)	Loss of "B" MG Set
4		C (RO) C(CRS)	Control Rod 22-35 inadvertently scrams (TS)
5		M (ALL)	Steam Leak from RCIC piping
6		C (BOP) C (CRS)	RCIC isolation valves fail to close
7		C (BOP) C (CRS)	"E" SRV fails ~~to close~~ open (open's T.S.)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: Hope Creek Scenario No.: 2 Op-Test No.: ______

Examiners: ______ Operators: ______

Initial Conditions: 80% power, middle of cycle. A control rod sequence exchange has just been performed and power is being raised back to 100%. The load dispatcher has requested a temporary hold at 80% power

Turnover:

SLC pump AP-208 has been tagged out for a motor replacement and is expected back in 48 hours. OPRM system is INOP due to an existing 10CFR21 issue. The OPRM system is still functional but is considered INOP per Tech Specs. No other equipment is Out of Service.
Maintain power at 80% until contacted by Load Dispatcher, then raise power to 100%.
HC.OP-ST.BE-0002, Core Spray Pump Loop A Full Flow Test is in progress and completed up to step 5.23, (pump testing). Complete HC.OP-ST.BE-0002 as soon as shift turnover is complete.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP) I (BOP) I (CRS)	Perform HC.OP-ST.BE-0002, Core Spray Pump Loop A Full flow test. Core Spray Loop "A" discharge flow instrument fails during full flow test
2		R (RO)	Power Increase Recirc Flow
3	BOP	~~I~~ (RO) I (CRS)	Inadvertent HPCI initiation
4		I (BOP) I (CRS)	~~Main Stack Rad. Monitor P/S~~ loss, FRVS fails to start TALK TO MIKE
5		C (~~BOP~~) (RO)	480 Volt Unit Substation 10B130 trips
6		M (ALL) I (RO)	EHC failure, Electrical ATWS PUMP TRIPS (SWAP PUMPS) (RRCS WORKS)
7	BOP	C (~~BOP~~)	HPCI Power supply Failure (DOESN'T MATTER)
8		M(ALL)	Broken SRV tailpipe, SRV Fails open, PSP function lost, ED required (NEED MORE - BROKEN DOWNCOMER)
9		I (BOP)	RHR Spray Logic failure (SWAP LOOPS)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: Hope Creek   Scenario No.: 3 (Spare)   Op-Test No.: ______

Examiners: ______________________   Operators: ______________________

Initial Conditions: Plant is at 80% power, middle of life, returning to power after a mini-outage, the operators are preparing to place the third RFP in service IAW HC.OP-SO.AE-0001. Severe weather is predicted for the upcoming shift.

Turnover:

Start the 3rd RFP and increase power to 100%

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP) N (CRS)	Start the 3rd RFP IAW HC.OP-SO.AE-0001
2		R (RO) R (CRS)	Commence load increase after starting RFP
3		I (RO) I (CRS)	"A" APRM fails
4		C (BOP) C (CRS)	Drywell Chiller Compressor fails
5		C (RO)	"B" Recirc Pump high vibration. Operator will trip
6		M (ALL) C (BOP)	Loss of off-site power due to storm "A" D/G ~~Tie~~ breaker failure to Auto Close (OUTPUT)
7		M (ALL)	Recirc suction piping leak. Small enough that crew can control parameters for loss of all high pressure feed
			* Scenario ends with Emergency Depressurization and level restored above TAF.

MOVE #7 TO SCENARIO #2 DOWNCOMER BREAK

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Facility: Hope Creek	Date of Examination: 11/28/05
Examination Level : RO	Operating Test Number: ______

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	S, A, N	Check Drywell to Torus D/P during power operations per Daily Surveillance Log
Conduct of Operations	N	Procedure Change - Make a change to a procedure for Emergent work
Equipment Control	D, A	Rod Worth Minimizer Operability -
Radiation Control	How N, A	Enter and exit a High Radiation Area for a valve lineup. 2.3.10
Emergency Plan		

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

* Type Codes & Criteria: (C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1), (A)lternate Path
(P)revious 2 exams (≤ 1; randomly selected)

Facility: Hope Creek	Date of Examination: 11/28/05
Exam Level (circle one): **RO**	Operating Test No.:

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

System / JPM Title	Type Code*	Safety Function
a. LPRM / Bypass failed LPRM (SE001) (using pics - not in simulator)	D, S	7
b. FRVS / Manually start FRVS system (GU001)	D, S	9
c. Recirc Flow Control System / Reset a Recirc MG Set Scoop Tube Lockup (Alt. Path - Recirc speed inexplicably rises following reset) (BB002)	M, A, S	1
d. Transfer station loads prior to shutting down Generator (Talk to Mike)	S, N, A	6
e. PCIS / Restart RWCU following Group Isolation How	N, A, S, E	5
f. Swap FW Level Control, Single to 3-Element will do something [illegible]	N, S	2
g. HPCI - Startup HPCI in the CST to CST mode (BJ002)	D, S, A	4
h. Main Steam System - Main Steam System recovery following a Group 1 isolation.	N, S, L	3

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

System / JPM Title	Type Code*	Safety Function
i. Main Steam/ Closing MSIV from Outside the Control Room	N, R	3
j. A/C Electrical / Startup a 20KVA Inverter	N, A	6
k. Control Rod Drive / Isolate a CRD HCU (BF006)	D, R	1

@ All control room (and in-plant) systems must be different and serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path (4)	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank (4)	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant (1)	1 / 1 / 1
(L)ow-Power (1)	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A) (7)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams (0)	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA (3)	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Hope Creek	Date of Examination: 11/28/05
Examination Level : SRO	Operating Test Number:

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	S, N, A	Determine SRM/IRM overlap per procedure and Tech Specs
Conduct of Operations	R, N	Procedure Change - Make a change to a procedure for Emergent work
Equipment Control	C, N, A	Review and approve a clearance prior to maintenance
Radiation Control	C, N, A	Enter and exit a High Radiation Area for a valve lineup. 2.3.10
Emergency Plan	S, M	Classify an Emergency Event - 2.4.41 - May be done after a scenario using the simulator.

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

* Type Codes & Criteria:
(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank ($\leq$ 3 for ROs; $\leq$ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank ($\geq$ 1), (A)lternate Path
(P)revious 2 exams ($\leq$ 1; randomly selected)

Facility: Hope Creek Date of Examination: 11/28/05
Exam Level (circle one): **SRO** Operating Test No.: ______

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

System / JPM Title	Type Code*	Safety Function
a. LPRM / Bypass failed LPRM (SE001)	D , S	7
b. FRVS / Manually start FRVS system (GU001)	D, S	9
c. Recirc Flow Control System / Reset a Recirc MG Set Scoop Tube Lockup (Alt. Path - Recirc speed inexplicably rises following reset) (BB002)	M, A, S	1
d. Transfer station loads prior to shutting down Generator	S, N, A	6
e. PCIS / Restart RWCU following Group Isolation	N, A, S, E	5
f. HPCI - Startup HPCI in the CST to CST mode (BJ002)	D, S, A	4
g. Main Steam System - Main Steam system recovery following a Group 1 isolation	N, S, L	3
h.		
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
i. Main Steam/ Closing MSIV from Outside the Control Room	N, R	3
j. A/C Electrical / Startup a 20KVA Inverter	N, R	6
k. Control Rod Drive / Remove an HCU from service (BF006)	D, R	1

@ All control room (and in-plant) systems must be different and serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path (4)	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank (4)	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant (1)	1 / 1 / 1
(L)ow-Power (1)	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A) (7)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams (0)	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA (3)	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Hope Creek Date of Exam: 11/28/05 Exam Level: RO ☒ SRO ☒

	Item Description							a	b*	c#
1.	Questions and answers are technically accurate and applicable to the facility.							MB	SP	SO
2.	a. NRC K/As are referenced for all questions. b. Facility learning objectives are referenced as available.							MB	SP	SO
3.	SRO questions are appropriate in accordance with Section D.2.d of ES-401							MB	SP	SO
4.	The sampling process was random and systematic (if more than 4 RO and 2 SRO questions are repeated from the last 2 NRC licensing exams, consult with NRR OL program office.)									SO
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	SP	SO
6.	Bank use meets limits (no more than 75 percent from the bank, at least 10 percent new, and the rest new or modified); enter the actual RO / SRO-only question distribution(s) at right.	Bank		Modified		New		MB	SP	SO
		23	5	27	5	25	15			
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		C/A				MB	SO	SO
		31	7	44	18					
8.	References/handouts provided do not give away answers or aid in the elimination of distractors.							MB	SO	SO
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	SP	SO
10.	Question psychometric quality and format meet the guidelines in ES Appendix B.							MB	SP	SO
11.	The exam contains the required number of one-point, multiple choice items; the total is correct and agrees with the value on the cover sheet.							MB	SO	SO

	Printed Name / Signature	Date
a. Author	Michael L. Brown / Michael L. Brown	10/4/05
b. Facility Reviewer (*)		
c. NRC Chief Examiner (#)	Steven Osmund / [signature]	10/4/05
d. NRC Regional Supervisor	(+) R.J. Conte / [signature]	10/5/05

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.

+ 50 old, 50 new for release - validation review + original to RO/HMS master file after exam

Facility: Hope Creek		Date of Examination: 11/28/05		
Item	Task Description	Initials a	b*	c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model, in accordance with ES-401.	MB	SD	SD
	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	SD	SD
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	MB	SD	SD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MB	SD	SD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.	MB	SD	SD
	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, and ensure that 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 that scenarios will not be repeated on subsequent days.	MB	SD	SD
	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	SD	SD
3. * W/T	a. Verify that the systems walk-through outline meets the criteria specified on Form ES-301-2: (1) the outline(s) contain(s) the required number of control room and in-plant tasks distributed among the safety functions as specified on the form (2) task repetition from the last two NRC examinations is within the limits specified on the form (3) no tasks are duplicated from the applicants' audit test(s) (4) the number of new or modified tasks meets or exceeds the minimums specified on the form (5) the number of alternate path, low-power, emergency, and RCA tasks meet the criteria on the form.	—	—	
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) the tasks are distributed among the topics as specified on the form (2) at least one task is new or significantly modified (3) no more than one task is repeated from the last two NRC licensing examinations	—	—	
	c. 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.	—	—	
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam sections.	MB	SD	SD
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.	MB	SD	SD
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	MB	SD	SD
	d. Check for duplication and overlap among exam sections.	MB	SD	SD
	e. Check the entire exam for balance of coverage.	MB	SD	SD
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	MB	SD	SD

	Printed Name/Signature	Date
a. Author	Michael L. Brown / Michael L. Brown	10/4/05
b. Facility Reviewer (*)		
c. NRC Chief Examiner (#)	Steven Dennis / [signature]	10/4/05
d. NRC Supervisor (+)	Rich Conte / [signature]	10/5/05

Note: # Independent NRC reviewer initial items in Column "c"; chief examiner concurrence required.

* THIS QA SHEET REFLECTS THE COMPLETE WRITTEN EXAM + SIM SCENARIO'S AS OF 10/4/05

(+) Original to NDRMS [illegible] RIZ after exam

Facility: **Hope Creek - RO Exam** Date of Exam: **11/28/2005**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			4	3	N/A		4	20			
	2	2	2	1				1	1			0	7			
	Tier Totals	5	5	4				5	4			4	27			
2. Plant Systems	1	2	2	3	3	2	3	3	2	2	2	2	26			
	2	2	1	1	1	1	1	1	2	1	1	0	12			
	Tier Totals	4	3	4	4	3	4	4	4	3	3	2	38			

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	3	2	2	3	10					

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

e. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7. * The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)								Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	1	0	0	0	0	0	AK1.03 - Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of Forced Core Flow Circulation: Thermal Limits :(CFR: 41.8 to 41.10 / 45.3)	3.6	1
295003 Partial or Complete Loss of AC / 6	0	0	0	0	1	0	AA2.05 - Ability to determine and interpret the following as they apply to Partial or Complete Loss of AC : Whether a partial or complete loss of A.C. Power has occurred:(CFR: 41.10 /43.5/ 45.13)	3.9	1
295004 Partial or Total Loss of DC Pwr / 6	0	0	1	0	0	0	AK3.01 - Knowledge of the reasons for the following responses as they apply to Partial or Total Loss of DC Pwr : Load shedding Plant Specific:(CFR: 41.5/41.10 / 45.6 /45.13)	2.6	1
295005 Main Turbine Generator Trip / 3	0	0	0	0	0	1	AG2.1.2 - Knowledge of operator responsibilities during all modes of plant operation (CFR: 41.10 / 45.13)	3.0	1
295006 SCRAM / 1	1	0	0	0	0	0	AK1.03 - Knowledge of the operational implications of the fallowing concepts as they apply to the SCRAM: Reactivity Control:(CFR: 41.8 to 41.10 /45.3)	3.7	1
295016 Control Room Abandonment / 7	0	0	0	0	0	1	AG2.1.30 - Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	3.9	1
295018 Partial or Total Loss of CCW / 8	0	0	0	0	1	0	AA2.04 - Ability to determine and interpret the following as they apply to Partial or Total Loss of CCW System Flow:(CFR: 41.10/43.5/ 45.13)	2.9	1
295019 Partial or Total Loss of Inst. Air / 8	0	0	0	1	0	0	AA1.03 - Ability to operate and / or monitor the following as they apply to Partial or Total Loss of Inst. Air: Instrument Air Compressor Power supplies:(CFR: 41.7145.5/45.6)	3.0	1
295021 Loss of Shutdown Cooling / 4	0	0	0	0	1	0	AA2.05 - Ability to determine and interpret the following as they apply to Loss of Shutdown Cooling: Reactor Vessel Metal Temperature (CFR: 41.10 /43.5/45.13)	3.4	1
295023 Refueling Acc / 8	0	1	0	0	0	0	AK2.03 - Knowledge of the interrelations between Refueling Accidents and the following: Radiation Monitoring equipment (CFR41.7 /45.7/ 45.8)	3.4	1
295024 High Drywell Pressure / 5	0	0	0	1	0	0	EA1.03 - Ability to operate and/ or monitor the following as they apply to High Drywell Pressure: LPCS - Plant specific (CFR41.7/ 45.5/ 45.6)	4.0	1
295025 High Reactor Pressure / 3	0	0	0	1	0	0	EA1.02 - Ability to operate and / or monitor the following as they apply to High Reactor Pressure : Reactor/Turbine pressure regulating system :(CFR: 41.7/45.5/ 45.6)	3.8	1

295026 Suppression Pool High Water Temp. / 5	0	0	0	0	0	1	EG2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 / 45.6)	3.9	1
295027 High Containment Temperature / 5	0	0	0	0	0	0			0
295028 High Drywell Temperature / 5	0	0	0	0	0	1	EG2.1.30 - Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	3.9	1
295030 Low Suppression Pool Wtr Lvl / 5	0	0	1	0	0	0	EK3.07 - Knowledge of the reasons for the following responses as they apply to Low Suppression Pool Wtr Lvl: NPSH considerations for ECCS pumps:(CFR: 41.5/41.10/45.6/ 45.13)	3.5	1
295031 Reactor Low Water Level / 2	0	1	0	0	0	0	EK2.10 - Knowledge of the interrelations between Reactor Low Water Level and the following: Redundant reactivity control: Plant specific (CFR: 41.7/45.7/45.8)	4.0	1
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	0	0	0	1	0	0	EA1.02 - Ability to operate and / or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: RRCS: Plant Specific (CFR: 41.7/45.5/ 45.6)	3.8	1
295038 High Off-site Release Rate / 9	0	0	1	0	0	0	EK3.02 - Knowledge of the reasons for the following responses as they apply to High Off-site Release Rate: System Isolations :(CFR: 41.5/41.10/45.6/ 45.13)	3.9	1
600000 Plant Fire On Site / 8	1	0	0	0	0	0	AK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Plant Fire On Site: Fire Classifications by type (CFR: 41.8 to 41.10 /45.3) ,	2.5	1
295005 Main Turbine Generator Trip / 3	0	1	0	0	0	0	AK2.04 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection (CFR: 41.7 / 45.8)	3.3	1
K/A Category Totals:	3	3	3	4	3	4	Group Point Total:		20

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)								Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Loss of Main Condenser Vacuum's Reactor Power - Plant Specific:(CFR: 41.10/43.5/ 45.13)	3.2	1
295007 High Reactor Pressure / 3	0	0	0	0	0	0			
295008 High Reactor Water Level / 2	0	0	1	0	0	0	AK3.06 - Knowledge of the reasons for the following responses as they apply to High Reactor Water Level: RCIC Turbine Trip - Plant Specific:(CFR: 41.5/41.10/ 45.6/45.13)	3.4	1
295009 Low Reactor Water Level / 2	1	0	0	0	0	0	AK1.02 - Knowledge of the operational implications of the following concepts as they apply to the Low Reactor Water Level: Recirculation pump net positive suction head: Plant Specific:(CFR: 41.8 to 41.10/45.3)	3.0	1
295010 High Drywell Pressure / 5	0	0	0	0	0	0			
295011 High Containment Temp / 5	0	0	0	0	0	0			
295012 High Drywell Temperature / 5	0	0	0	0	0	0			
295013 High Suppression Pool Temp. / 5	0	0	0	0	0	0			
295014 Inadvertent Reactivity Addition / 1	0	0	0	0	0	0			
295015 Incomplete SCRAM / 1	0	0	0	0	0	0			
295017 High Off-site Release Rate / 9	0	0	0	0	0	0			
295020 Inadvertent Cont. Isolation / 5 & 7	0	0	0	0	0	0			
295022 Loss of CRD Pumps / 1	0	0	0	0	0	0			
295029 High Suppression Pool Wtr Lvl / 5	0	1	0	0	0	0	EK2.07 - Knowledge of the interrelations High Suppression Pool Wtr Lvl and the following: Drywell/ containment water level:(CFR: 41.7 /45.7/45.8)	3.1	1
295032 High Secondary Containment Area Temperature / 5	0	0	0	0	0	0			
295033 High Secondary Containment Area Radiation Levels / 9	0	0	0	0	0	0			

295034 Secondary Containment Ventilation High Radiation / 9	0	0	0	1	0	0	EA1.01 - Ability to operate and/ or monitor the following as they apply to Secondary Containment Ventilation High Radiation: Area radiation monitoring system:(CFR41.7/45.5/45.6)	3.8	1
295035 Secondary Containment High Differential Pressure / 5	0	0	0	0	0	0			0
295036 Secondary Containment High Sump/Area Water Level / 5	1	0	0	0	0	0	EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Secondary Containment High Sump/ Area Water Level: Radiation releases (CFR:41.8 to 41.10/45.3)	2.9	1
500000 High CTMT Hydrogen Conc. / 5	0	1	0	0	0	0	EK2.02 - Knowledge of the interrelations between High CTMT Hydrogen Conc. And the following: Containment oxygen monitoring systems (CFR: 41.7 / 45.7 /45.8)	3.1	1
K/A Category Point Totals:	2	2	1	1	1	0	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 1 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode	0	0	0	0	0	0	1	0	0	0	0	A1.04 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: System Pressure (CFR: 41.5 / 45.5)	2.5	1
205000 Shutdown Cooling	0	0	0	0	0	0	0	0	1	0	0	A3.03 - Ability to monitor automatic operations of the Shutdown Cooling System(RHR Shutdown Cooling Mode) including: lights and alarms (CFR:41.7/45.5)	3.5	1
206000 HPCI	0	0	0	0	0	0	0	1	0	0	0	A2.07 - Ability to (a) predict the impacts of the following on the HPCI and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Low suppression pool level: BWR-2, 3, 4 (CFR:41.5/43.5/45.3/45.13)	3.4	1
206000 HPCI	0	0	0	0	1	0	0	0	0	0	0	K5.05 - Knowledge of the operational implications of the following concepts as they apply to the HPCI: Turbine speed control: BWR- 2,3,4 (CFR:41.5/ 45.7)	3.3	1
207000 Isolation (Emergency) Condenser	0	0	0	0	0	0	0	0	0	0	0			
209001 LPCS	0	1	0	0	0	0	0	0	0	0	0	K2.01 - Knowledge of electrical power supplies to the following: Pump power (CFR41.7)	3.0	1
209002 HPCS	0	0	0	0	0	0	0	0	0	0	0			
211000 SLC	0	0	0	1	0	0	0	0	0	0	0	K4.04 - Knowledge of SLC design feature(s) and or interlock(s) which provide for the following: Indication of fault in explosive valve firing circuits (CFR41.7)	3.8	1
212000 RPS	0	0	1	0	0	0	0	0	0	0	0	K3.11 - Knowledge of the effect that a loss or malfunction of the RPS will have on the following: Recirculation system (CFR41.7/45.6)	3.0	1
215003 IRM	0	0	0	1	0	0	0	0	0	0	0	K4.04 - Knowledge of the IRM design feature(s) and or interlock(s) which provide for the following: Varying system sensitivity levels using range switches (CFR41.7)	2.9	1
215003 IRM	0	1	0	0	0	0	0	0	0	0	0	K2.01 - Knowledge of electrical power supplies to the following: IRM Channels/ detectors (CFR41.7)	2.5	1

215004 Source Range Monitor	1	0	0	0	0	0	0	0	0	0	0	K1.02- Knowledge of the physical connections and/or cause-effect relationships between Source Range Monitor and the following: Reactor Manual Control (CFR:41.2 to 41.9/45.7 to 45.8)	3.4	1
215005 APRM / LPRM	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purposes and function of major system components and controls (CFR: 41.7)	3.2	1
217000 RCIC	1	0	0	0	0	0	0	0	0	0	0	K1.01 - Knowledge of the physical connections and/or cause-effect relationships between RCIC and the following: Condensate storage and transfer system (CFR:41.2 to 41.9/ 45.7 to 45.8)	3.5	1
218000 ADS	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purpose and function of major system components and controls.	3.2	1
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	0	0	0	0	1	0	A4.02 - Ability to manually operate and/or monitor in the control room: Manually initiate the system (CFR:41.7/45.5 to 45.8)	3.9	1
239002 SRVs	0	0	0	0	0	0	0	0	0	1	0	A4.06- Ability to manually operate and/or monitor in the control room: Reactor water level (CFR: 41.7/45.5 to 45.8)	3.9	1
259002 Reactor Water Level Control	0	0	1	0	0	0	0	0	0	0	0	K3.06 - Knowledge of the effect that a loss or malfunction of the Reactor Water Level Control will have on the following: Main Turbine (CFR:41.7/45.6)	2.8	1
261000 SGTS	0	0	1	0	0	0	0	0	0	0	0	K3.02 - Knowledge of the effect that a loss or malfunction of the SGTS will have on the following: Off-site release rate (CFR:41.7/45.6)	3.6	1
262001 AC Electrical Distribution	0	0	0	1	0	0	0	0	0	0	0	K4.03 - Knowledge of AC Electrical distribution design feature(s) and or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers (CFR:41.7)	3.1	1
262002 UPS (AC/DC)	0	0	0	0	0	1	0	0	0	0	0	K6.02 - Knowledge of the effect that a loss or malfunction of the following will have on the UPS (AC/DC): DC electrical power (CFR:41.7/45.7)	2.8	1
263000 DC Electrical Distribution	0	0	0	0	0	0	1	0	0	0	0	A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical distribution controls including: Battery charging/discharging rate (CFR:41.5/45.5)	2.5	1
264000 EDGs	0	0	0	0	0	0	0	1	0	0	0	A2.04 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Consequences of operating under/over excited (CFR: 41.5 / 45.6)	2.9	1
300000 Instrument Air	0	0	0	0	0	0	0	0	1	0	0	A3.02 - Ability to monitor automatic operations of the Instrument Air including: Air temperature (CFR 41.7/45.5)	2.9	1
262001 A.C. Electrical Distribution	0	0	0	0	1	0	0	0	0	0	0	K5.02 - Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Breaker Control (CFR: 41.5 / 45.3)	2.6	1

400000 Component Cooling Water	0	0	0	0	0	1	0	0	0	0	0	K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the Component Cooling Water: Valves (CFR:41.5/45.5)	2.7	1
215004 Source Range Monitor	0	0	0	0	0	0	1	0	0	0	0	A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: RPS status (CFR: 41.5 / 45.5)	3.4	1
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	1	0	0	0	0	0	K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF :Nuclear boiler instrumentation (CFR: 41.7 / 45.7)	3.3	1
K/A Category Point Totals:	2	2	3	3	2	3	3	2	2	2	2	Group Point Total:		26

ES-401												BWR Examination Outline Plant Systems - Tier 2/Group 2 (RO / SRO)	Form ES-401-1	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic	0	0	0	0	0	0	0	0	0	0	0			
201002 RMCS	0	0	0	0	0	0	0	0	0	0	0			
201003 Control Rod and Drive Mechanism	0	0	0	0	0	0	0	0	0	0	0			
201004 RSCS	0	0	0	0	0	0	0	0	0	0	0			
201005 RCIS	0	0	0	0	0	0	0	0	0	0	0			
201006 RWM	0	0	0	0	0	1	0	0	0	0	0	K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the RWM: Rod Position indication - Plant Specific	2.9	1
202001 Recirculation	0	0	0	0	0	0	0	0	0	0	0			
202002 Recirculation Flow Control	0	0	0	0	0	0	0	1	0	0	0	A2.07 - Ability to (a) predict the impacts of the following on the Recirculation flow control and (b) based on those predications, use procedures to correct, control, or mitigate the conseqences of those abnormal operation: Loss of feedwater singal inputs: Plant specific (CFR:41.5/43.5/45.3/45.13)	3.3	1
204000 RWCU	0	0	0	0	0	0	0	0	0	0	0			
214000 RPIS	0	0	0	0	0	0	0	0	0	0	0			
215001 Traversing In-core Probe	0	0	0	0	0	0	0	0	0	0	0			
215002 RBM	0	0	0	0	0	0	0	0	0	0	0			
216000 Nuclear Boiler Inst.	0	0	0	0	0	0	0	0	0	0	0			
219000 RHR/LPCI: Torus/Pool Cooling Mode	0	0	0	1	0	0	0	0	0	0	0	K4.03 - Knowledge of RHR/LPCI Torus/Pool Cooling Mode design feature(s) and or interlocks which provide for the following: Unintentional reduction in vessel injection flow during accident conditions: plant specific (CFR:41.7)	3.8	1
223001 Primary CTMT and Aux.	0	1	0	0	0	0	0	0	0	0	0	K2.09 - Knowledge of electrical power supplies to the following: Drywell cooling fans: Plant-Specific (CFR: 41.7)	2.7	1
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
230000 RHR/LPCI: Torus/Pool Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
233000 Fuel Pool Cooling/Cleanup	0	0	0	0	0	0	0	0	0	0	0			
234000 Fuel Handling Equipment	0	0	0	0	0	0	0	0	0	0	0			
239001 Main and Reheat Steam	0	0	0	0	0	0	0	0	1	0	0	A3.01 - Ability to monitor automatic operations of the Main and Reheat system including: Isolation of main steam system (CFR:41.7/45.5)	4.2	1

239003 MSIV Leakage Control	0	0	0	0	0	0	0	0	0	0	0			
241000 Reactor/Turbine Pressure Regulator	0	0	0	0	0	0	0	0	0	0	0			
245000 Main Turbine Gen. / Aux.	1	0	0	0	0	0	0	0	0	0	0	K1.02 - Knowledge of the physical connections and/or cause effect relationships between Main Turbine Generator / Aux and the following: Condensate system (CFR:41.2 to 41.9 / 45.7 to 45.8)	2.5	1
256000 Reactor Condensate	0	0	0	0	0	0	0	0	0	0	0			
259001 Reactor Feedwater	0	0	0	0	0	0	0	0	0	0	0			
268000 Radwaste	0	0	0	0	0	0	0	1	0	0	0	A2.01 - Ability to (a) predict the impacts of the following on the Radwaste and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal operation: System rupture (CFR:41.5/ 43.5/ 45.3/ 45.13)	2.9	1
271000 Offgas	0	0	0	0	0	0	0	0	0	0	0			
272000 Radiation Monitoring	0	0	0	0	1	0	0	0	0	0	0	K5.01 - Knowledge of the operational implications of the following concepts as they apply to the Radiation Monitoring: Hydrogen injection operation's effect on process radiation indications: Plant specific (CFR: 41.5/ 45.7)	3.2	1
286000 Fire Protection	0	0	0	0	0	0	0	0	0	0	0			
288000 Plant Ventilation	0	0	0	0	0	0	0	0	0	0	0			
290001 Secondary CTMT	0	0	0	0	0	0	0	0	0	0	0			
290003 Control Room HVAC	0	0	0	0	0	0	1	0	0	0	0	A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the Control Room HVAC controls including: Area Temperatures (CFR41.5/45.5)	2.6	1
290002 Reactor Vessel Internals	1	0	0	0	0	0	0	0	0	0	0	K1.20 - Knowledge of the physical connections and/or cause effect relationships between Reactor Vessel Internals and the following: Nuclear Instrumentation (CFR:41.2 to 41.9/ 45.7 to 45.8)	3.2	1
223001 Primary CTMT and Aux.	0	0	0	0	0	0	0	0	0	1	0	A4.12 - Ability to manually operate and/or monitor in the control room: Drywell coolers/chillers (CFR: 41.7 / 45.5 to 45.8)	3.5	1
223001 Primary CTMT and Aux.	0	0	1	0	0	0	0	0	0	0	0	K3.01 - Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will *have on following: Secondary containment* (CFR: 41.7 / 45.4)	3.6	1
K/A Category Point Totals:	2	1	1	1	1	1	1	2	1	1	0	Group Point Total:		12

Facility: Hope Creek - **RO Exam** Date of Exam: **11/28/05**

Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.21	Ability to obtain and verify controlled procedure copy (CFR: 45.10 / 45.13)	3.1	1		
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel (CFR: 43.5 / 45.12)	2.5	1		
	2.1.33	Ability to recognize indications for system operating parameters which are entry-level condition for Technical Specifications (CFR: 43.2 / 43.3 / 45.3)	3.4	1		
	2.1.					
	2.1.					
	Subtotal			3		
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1)	3.7	1		
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity (CFR: 43.6)	2.8	1		
	2.2.					
	Subtotal			2		
3. Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements (CFR: 41.12 / 43.4. 45.9 / 45.10).	2.6	1		
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure (CFR: 43.4 / 45.10)	2.9	1		
	2.3.					
	Subtotal			2		

4. Emergency Procedures/ Plan	2.4.27	Knowledge of fire in the plant procedure (CFR: 41.10 / 43.5 / 45.13)	3.0	1		
	2.4.39	Knowledge of the RO's responsibilities in emergency plan implementation (CFR: 45.11)	3.3	1		
	2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3)	3.3	1		
	2.4.					
	Subtotal			3		
Tier 3 Point Total				10		

<table>
<tr><td colspan="18">Facility: Hope Creek - SRO Only Exam Date of Exam: 11/28/05</td></tr>
<tr><th rowspan="2">Tier</th><th rowspan="2">Group</th><th colspan="12">RO K/A Category Points</th><th colspan="4">SRO-Only Points</th></tr>
<tr><th>K 1</th><th>K 2</th><th>K 3</th><th>K 4</th><th>K 5</th><th>K 6</th><th>A 1</th><th>A 2</th><th>A 3</th><th>A 4</th><th>G *</th><th>Total</th><th colspan="2">A2</th><th>G*</th><th>Total</th></tr>
<tr><td rowspan="3">1. Emergency & Abnormal Plant Evolutions</td><td>1</td><td></td><td></td><td></td><td colspan="3" rowspan="3">N/A</td><td></td><td></td><td colspan="2" rowspan="3">N/A</td><td></td><td></td><td colspan="2">3</td><td>4</td><td>7</td></tr>
<tr><td>2</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">2</td><td>1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">5</td><td>5</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">3</td><td>2</td><td>5</td></tr>
<tr><td>2</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">1</td><td>2</td><td>3</td></tr>
<tr><td>Tier Totals</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2">4</td><td>4</td><td>8</td></tr>
<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2"></td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
<tr><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td><td>2</td><td>2</td><td>1</td><td>2</td></tr>
</table>

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

e. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. * The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	0	0	0	0	0	0			
295003 Partial or Complete Loss of AC / 6	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295004 Partial or Total Loss of DC Pwr / 6	0	0	0	0	1	0	AA2.04 - Ability to determine and interpret the following as they apply to Partial or Total loss of DC power:(CFR: 41.10 J 43.5 / 45.13) - System Lineups	3.3	1
295005 Main Turbine Generator Trip / 3	0	0	0	0	0	0	K/A Randomly Rejected		
295006 SCRAM / 1	0	0	0	0	0	1	AG2.1.32 - Ability to explain and apply system limits and precautions (CFR 41.10/ 43.2/ 45.12)	3.8	1
295016 Control Room Abandonment / 7	0	0	0	0	0	0			
295018 Partial or Total Loss of CCW / 8	0	0	0	0	0	1	G2.4.30 - Knowledge of which events related to system operations/status should be reported to outside agencies	3.6	1
295019 Partial or Total Loss of Inst. Air / 8	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Partial or Total loss of Instrument Air:(CFR: 41.10/43,5/ 45.13) - Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	3.7	1
295021 Loss of Shutdown Cooling / 4	0	0	0	0	0	0			
295023 Refueling Acc / 8	0	0	0	0	0	0			
295024 High Drywell Pressure / 5	0	0	0	0	0	0			
295025 High Reactor Pressure / 3	0	0	0	0	0	0			
295026 Suppression Pool High Water Temp. / 5	0	0	0	0	0	0			
295027 High Containment Temperature / 5	0	0	0	0	0	0			
295028 High Drywell Temperature / 5	0	0	0	0	0	1	EG2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR 45.3)	3.3	1
295030 Low Suppression Pool Wtr Lvl / 5	0	0	0	0	1	0	EA2.01 - Ability to determine and interpret the following as they apply to Low Suppression Pool Water level (CFR:41.10/ 43.5/ 45.13) - Suppression Pool level	4.2	1

295031 Reactor Low Water Level / 2	0	0	0	0	0	0			
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	0	0	0	0	0	0			
295038 High Off-site Release Rate / 9	0	0	0	0	0	0			
600000 Plant Fire On Site / 8	0	0	0	0	0	0			
K/A Category Totals:					3	4	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	0	0	0	0	0	0			
295007 High Reactor Pressure / 3	0	0	0	0	0	0			
295008 High Reactor Water Level / 2	0	0	0	0	0	0			
295009 Low Reactor Water Level / 2	0	0	0	0	1	0	AA2.02 - Ability to determine and interpret the following as they apply to Low Reactor Water Level (CFR: 41.10/ 43.5 / 45.13) - Steam flow/ feed flow mismatch	3.7	1
295010 High Drywell Pressure / 5	0	0	0	0	0	1	AG2.4.6 - Knowledge of symptom based EOP mitigation strategies (CFR: 41.10 / 43.5 / 45.13)	4.0	1
295011 High Containment Temp / 5	0	0	0	0	0	0			
295012 High Drywell Temperature / 5	0	0	0	0	1	0	AA2.01 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell temperature (CFR: 41.10 / 43.5 / 45.13)	3.8	1
295013 High Suppression Pool Temp. / 5	0	0	0	0	0	0			
295014 Inadvertent Reactivity Addition / 1	0	0	0	0	0	0			
295015 Incomplete SCRAM / 1	0	0	0	0	0	0			
295017 High Off-site Release Rate / 9	0	0	0	0	0	0			
295020 Inadvertent Cont. Isolation / 5 & 7	0	0	0	0	0	0			
295022 Loss of CRD Pumps / 1	0	0	0	0	0	0			
295029 High Suppression Pool Wtr Lvl / 5	0	0	0	0	0	0			
295032 High Secondary Containment Area Temperature / 5	0	0	0	0	0	0			
295033 High Secondary Containment Area Radiation Levels / 9	0	0	0	0	0	0			
295034 Secondary Containment Ventilation High Radiation / 9	0	0	0	0	0	0			

295035 Secondary Containment High Differential Pressure / 5	0	0	0	0	0	0			
295036 Secondary Containment High Sump/Area Water Level / 5	0	0	0	0	0	0			
500000 High CTMT Hydrogen Conc. / 5	0	0	0	0	0	0			
K/A Category Point Totals:					2	1	Group Point Total:		3

ES-401	BWR Examination Outline Plant Systems - Tier 2/Group 1 (RO / SRO)	Form ES-401-1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode	0	0	0	0	0	0	0	0	0	0	0			
205000 Shutdown Cooling	0	0	0	0	0	0	0	0	0	0	0			
206000 HPCI	0	0	0	0	0	0	0	0	0	0	1	G2.1.14 - Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5 / 45.12)	3.3	1
207000 Isolation (Emergency) Condenser	0	0	0	0	0	0	0	0	0	0	0			
209001 LPCS	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the LPCS and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Valve closures	3.2	1
209002 HPCS	0	0	0	0	0	0	0	0	0	0	0			
211000 SLC	0	0	0	0	0	0	0	0	0	0	0			
212000 RPS	0	0	0	0	0	0	0	0	0	0	0			
215003 IRM	0	0	0	0	0	0	0	0	0	0	0			
215004 Source Range Monitor	0	0	0	0	0	0	0	0	0	0	0			
215005 APRM / LPRM	0	0	0	0	0	0	0	1	0	0	0	A2.02 - Ability to (a) predict the impacts of the following on the APRM/ LPRM and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Upscale or downscale trips.	3.7	1
217000 RCIC	0	0	0	0	0	0	0	0	0	0	0			
218000 ADS	0	0	0	0	0	0	0	0	0	0	0			
223002 PCIS/Nuclear Steam Supply Shutoff	0	0	0	0	0	0	0	0	0	0	0			
239002 SRVs	0	0	0	0	0	0	0	0	0	0	0			
259002 Reactor Water Level Control	0	0	0	0	0	0	0	0	0	0	1	G2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3)	4.0	1

261000 SGTS	0	0	0	0	0	0	0	0	0	0	0			
262001 AC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
262002 UPS (AC/DC)	0	0	0	0	0	0	0	0	0	0	0			
263000 DC Electrical Distribution	0	0	0	0	0	0	0	0	0	0	0			
264000 EDGs	0	0	0	0	0	0	0	1	0	0	0	A2.08 - Ability to (a) predict the impacts of the following on the EDGs and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Initiation of emergency generator room fire protection system.	3.7	1
300000 Instrument Air	0	0	0	0	0	0	0	0	0	0	0			
400000 Component Cooling Water	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 2 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic	0	0	0	0	0	0	0	0	0	0	1	G2.1.28 - Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)	3.3	1
201002 RMCS	0	0	0	0	0	0	0	0	0	0	0			
201003 Control Rod and Drive Mechanism	0	0	0	0	0	0	0	0	0	0	0			
201004 RSCS	0	0	0	0	0	0	0	0	0	0	0			
201005 RCIS	0	0	0	0	0	0	0	0	0	0	0			
201006 RWM	0	0	0	0	0	0	0	0	0	0	0			
202001 Recirculation	0	0	0	0	0	0	0	0	0	0	1	G2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 45.2 /45.6)	4.0	1
202002 Recirculation Flow Control	0	0	0	0	0	0	0	0	0	0	0			
204000 RWCU	0	0	0	0	0	0	0	0	0	0	0			
214000 RPIS	0	0	0	0	0	0	0	0	0	0	0			
215001 Traversing In-core Probe	0	0	0	0	0	0	0	0	0	0	0			
215002 RBM	0	0	0	0	0	0	0	0	0	0	0			
216000 Nuclear Boiler Inst.	0	0	0	0	0	0	0	0	0	0	0			
219000 RHR/LPCI: Torus/Pool Cooling Mode	0	0	0	0	0	0	0	0	0	0	0			
223001 Primary CTMT and Aux.	0	0	0	0	0	0	0	0	0	0	0			
226001 RHR/LPCI: CTMT Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
230000 RHR/LPCI: Torus/Pool Spray Mode	0	0	0	0	0	0	0	0	0	0	0			
233000 Fuel Pool Cooling/Cleanup	0	0	0	0	0	0	0	0	0	0	0			
234000 Fuel Handling Equipment	0	0	0	0	0	0	0	0	0	0	0			
239001 Main and Reheat Steam	0	0	0	0	0	0	0	0	0	0	0			
239003 MSIV Leakage Control	0	0	0	0	0	0	0	0	0	0	0			
241000 Reactor/Turbine Pressure Regulator	0	0	0	0	0	0	0	0	0	0	0			

245000 Main Turbine Gen. / Aux.	0	0	0	0	0	0	0	1	0	0	0	A2.05 - Ability to (a) predict the impacts of the following on the Main Turbine Gen. / Aux and (b) based on those predictions, use procedures to correct control or mitigate the consequences of those abnormal operation (CFR: 41.5/ 43.5/ 45.3/ 45.13) - Generator trip	3.8	1
256000 Reactor Condensate	0	0	0	0	0	0	0	0	0	0	0			
259001 Reactor Feedwater	0	0	0	0	0	0	0	0	0	0	0			
268000 Radwaste	0	0	0	0	0	0	0	0	0	0	0			
271000 Offgas	0	0	0	0	0	0	0	0	0	0	0			
272000 Radiation Monitoring	0	0	0	0	0	0	0	0	0	0	0			
286000 Fire Protection	0	0	0	0	0	0	0	0	0	0	0			
288000 Plant Ventilation	0	0	0	0	0	0	0	0	0	0	0			
290001 Secondary CTMT	0	0	0	0	0	0	0	0	0	0	0			
290003 Control Room HVAC	0	0	0	0	0	0	0	0	0	0	0			
290002 Reactor Vessel Internals	0	0	0	0	0	0	0	0	0	0	0			
K/A Category Point Totals:								1			2	Group Point Total:		3

Facility: **Hope Creek - SRO Only Exam** Date of Exam: **11/28/05**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13)			4.4	1
	2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits (CFR: 41.10 / 43.5 / 45.12)			2.9	1
	Subtotal					2
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities (CFR: 43.5 / 45.13)			3.3	1
	2.2.21	Knowledge of pre- and post-maintenance operability requirements (CFR: 43.2)			3.5	1
	Subtotal					2
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)			3.1	1
	Subtotal					1
4. Emergency Procedures/ Plan	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 43.5 / 45.12)			4.0	1
	2.4.36	Knowledge of chemistry/health physics tasks during emergency operations (CFR: 43.5)			2.8	1
	2.4.					
	Subtotal					2
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
Tier 1/ Group 1 RO exam	295027 EK2.01	K/A is for a Mark III containment and Hope Creek has a Mark I containment
Tier 2/ Group 1 RO exam	259002, A1.06	Hope Creek does not have (FWCI) Feedwater Coolant Injection
Tier 2/ Group 1 RO exam	262002, A1.02	Not applicable to Hope Creek
Tier 2/ Group 2 RO exam	215002, A4.04	Not applicable to Hope Creek
Tier 2/ Group 2 RO exam	223001 A4.02	Not applicable to Hope Creek
Tier 3 RO exam	G2.2.3	Not applicable to Hope Creek - Not a Multi-unit facility
Tier 2/ Group 1 RO exam	203000 G2.2.25	RO's not required to know bases
Tier 2 Group 1 RO exam	217000 K1.05	Connection between RCIC/RHR no longer used
Tier 2 Group 1 RO exam	218000 A2.04	ADS is always inhibited, therefore there is NO effect on a failure of ADS to initiate
Tier 2 Group 1 RO exam	264000 G2.1.14	RO's not required to make notifications on EDGs
Tier 2 Group 1 RO exam	300000 K5.01	Too many Instrument Air Questions
Tier 2 Group 2 RO exam	286000 K2.03	Too many Fire Protection Questions
Tier 2 Group 2 RO exam	226001 A4.14	No relationship between Containment Spray and Suppression Pool temperature

Tier 1 Group 1 SRO exam	600000 AA2.13	Too many fire protection questions
Tier 1 Group 2 SRO exam	295035 EG2.4.6	No EOP for Secondary Containment High differential pressure

[illegible]

REACTOR OPERATOR EXAM

Lower Cognitive Level ==>	31	Bank	23	30.7%	of 75 Q's
Higher Cognitive Level ==>	44	Mod	27	36.0%	of 75 Q's
Percent Higher Cognitive ==>	58.7%	New	25	33.3%	of 75 Q's

(RO Goal: 50% to 60% Higher Order) (Max 75% From Bank)

Choice A ==>	22	Choice C ==>	14
Choice B ==>	19	Choice D ==>	20

RO Questions Complete ==> 75 of 75 Q's

RO Exam Percent Complete ==> 100.0%

RO Questions With Handouts ==> 9

[illegible]

[illegible]	7	[illegible]	5	20.0%	of 25 Q's
[illegible]	18	[illegible]	5	20.0%	of 25 Q's
[illegible]	72.0%	[illegible]	15	60.0%	of 25 Q's

(Max 75% From Bank)

[illegible]	6	[illegible]	6
[illegible]	10	[illegible]	3

SRO Questions Complete ==> 25 of 25 Q's

SRO Exam Percent Complete ==> 100.0%

SRO Questions With Handouts ==> 11

[illegible]

REACTOR OPERATOR EXAM

Lower Cognitive Level ==>	31	Bank	23	30.7%	of 75 Q's
Higher Cognitive Level ==>	44	Mod	27	36.0%	of 75 Q's
Percent Higher Cognitive ==>	58.7%	New	25	33.3%	of 75 Q's

(RO Goal: 50% to 60% Higher Order) (Max 75% From Bank)

Choice A ==>	22	Choice C ==>	14
Choice B ==>	19	Choice D ==>	20

RO Questions Complete ==> 75 of 75 Q's

Facility: Hope Creek Date of Exam: 11/28/05 Scenario Numbers: 1/2/3 Operating Test No.:

	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		SD
2.	The scenarios consist mostly of related events.	MB		SD
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		SD
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		SD
5.	The events are valid with regard to physics and thermodynamics.	MB		SD
6.	Sequencing and timing of events is reasonable, and allows the examination team to obtain complete evaluation results commensurate with the scenario objectives.	MB		SD
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 constraints. Cues are given.	N/A		SD
8.	The simulator modeling is not altered.	MB		SD
9.	The scenarios have been validated. Pursuant to 10 CFR 55.46(d), any open simulator performance deficiencies or deviations from the referenced plant have been evaluated to ensure that functional fidelity is maintained while running the planned scenarios.	~~MB~~		~~SD~~
10.	Every operator will be evaluated using at least one new or significantly modified scenario. All other scenarios have been altered in accordance with Section D.5 of ES-301.	MB		SD
11.	All individual operator competencies can be evaluated, as verified using Form ES-301-6 (submit the form along with the simulator scenarios).	MB		SD
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		SD
13.	The level of difficulty is appropriate to support licensing decisions for each crew position.	MB		SD

	Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes	--	--	--
1.	Total malfunctions (5–8)	6 / 7 / 5	MB		SD
2.	Malfunctions after EOP entry (1–2)	2 / 1 / 1	MB		SD
3.	Abnormal events (2–4)	3 / 4 / 3	MB		SD
4.	Major transients (1–2)	1 / 1 / 1	MB		SD
5.	EOPs entered/requiring substantive actions (1–2)	1 / 2 / 1	MB		SD
6.	EOP contingencies requiring substantive actions (0–2)	1 / 1 / 0	MB		SD
7.	Critical tasks (2–3)	3 / 2 / 2	MB		SD

By Conte 10/5/05 for release to facility validation
Note essential elements of ESD-1 & 02 need to be specified

Facility: Hope Creek							Date of Exam: 11/28/05					Operating Test No.:			
APPLICANT	EVENT TYPE	Scenarios													
		1			2			3			4			TOTAL	MINIMUM
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION				
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		
RO	RX		1											1	1*
	NOR						1							1	1*
	I/C		2,3,4				1,4,5, 9							7	4*
	MAJ		5				6,8							3	2
	TS														2
SRO-I	RX	1				2								2	1*
	NOR													0	1*
	I/C	2,3,4, 5,6				3,6, 7								8	4*
	MAJ	5				6,8								3	2
	TS	3,4												2	2
RO	RX		1											1	1*
	NOR						1							1	1*
	I/C		2,3 , 4				1,4,5, 9							7	4*
	MAJ		5				6,8							3	2
	TS														2
SRO-I	RX	1				2								2	1*
	NOR													0	1*
	I/C	2,3,4, 5,6				3,6, 7								8	4*
	MAJ	5				6,8								3	2
	TS	3,4												2	2

Instructions:

1. Circle the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls (ATC)" and "balance-of-plant (BOP)" positions; Instant SROs must do one scenario, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position.

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 1-for-1 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: Michael E. Brown

NRC Reviewer:

HOPE CREEK ELECTRIC GENERATING STATION
NRC INITIAL LICENSED EXAMINATION SCENARIO 1
NOVEMBER 28, 2005

SCENARIO TITLE: **Reactor Startup/ Loss of 1BY160/ Steam Leak**

SCENARIO NUMBER: **NRC-001**

EFFECTIVE DATE:

EXPECTED DURATION: 1.0 Hours

REVISION NUMBER: **00**

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **OTHER** ____________________

REVISION SUMMARY:

New Scenario.

PREPARED BY: M. L. Brown — 9/29/05

NRC Operations Examiner — **DATE**

FACILITY REVIEWER: ____________________ ____________

Nuclear Operations Training Supervisor – Hope Creek — **DATE**

APPROVED BY: ____________________ ____________

NRC Chief Examiner — **DATE**

I. OBJECTIVE(S):

Enabling Objectives

A. The crew must demonstrate the ability to operate effectively as a team while completing a series of CREW CRITICAL TASKS, which measure the crew's ability to safely operate the plant during normal, abnormal, and emergency plant conditions.
(Crew critical tasks within this examination scenario guide are identified with an " *.")

II. MAJOR EVENTS:

A. Withdraw Control Rods until 1 bypass valve is open
B. CRD Flow Controller fails downscale in AUTO
C. Loss of "B" MG Set
D. Control Rod 22-35 inadvertently scrams (TS)
E. Steam Leak from RCIC piping
F. RCIC isolation valves fail to close
G. "E" SRV fails Open

III. SCENARIO SUMMARY:

The scenario begins with a Reactor Startup in service with IOP-3 completed up to step 5.3.28. Reactor power is approximately 4%. RCIC is being operated for HC.OP-ST.BD-0002, RCIC Pump Valve and Flow Test and should be completed within the next hour. 1BP116 EHC pump is tagged out for maintenance and will be out of service until a new pressure compensator arrives tomorrow. After the operators have raised power using the control rods, the CRD flow controller fails downscale in AUTO and the operators are forced to take MANUAL control of the flow controller. Once the operators have stabilized the plant, the "B" MG Set trips causing a ½ scram and RWCU isolation. The operators will have to restore power to the "B" RPS from the alternate source. Once power has been restored to the "B" RPS bus, Control Rod 22-35 inadvertently partially scrams (stops at position 12) and its Accumulator is INOP. With Reactor Pressure < 900 psig and Charging Header pressure < 940 psig, this will require a Scram. After the crew has stabilized from the scram, a steam leak develops on RCIC, RCIC isolation valves fail to close causing RCIC room temperature to increase. Crew should enter EOP-103 based on high room temperature and place FRVS in service and attempt to shutdown RCIC. Crew should discover that the RCIC isolation valves can't be closed. Crew should scram the reactor based upon RCIC room temperature approaching safe operating limit and enter EOP-101. When temperature exceeds Max Safe Operation limit in 2 areas, crew should enter EOP-202, Emergency Depressurization. When the crew goes to Emergency Depressurize, "E" ADS valve will not open, requiring the BOP to open another SRV. Scenario will end after 5 SRVs have been opened.

IV. INITIAL CONDITIONS:

IC

Initial	
______	**INITIALIZE** the simulator to 4% power, MOL
______	
______	

PREP FOR TRAINING (i.e., RM11 set points, procedures, bezel covers)

Initial	Description
______	**COMPLETE** Attachment 2 "Simulator Ready-for-Training/Examination Checklist" of NC.TQ-DG.ZZ-0002(Z).

EVENT TRIGGERS:

Initial	ET #	Description	
	1	EVENT ACTION: COMMAND: PURPOSE:	
	2	EVENT ACTION: COMMAND: PURPOSE:	
	3	EVENT ACTION: COMMAND: PURPOSE:	

MALFUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	RT-1	---	100%
		---	---	RT-2	---	---
		---	---	RT-3	---	---
		4 min	15 min	ET-1	0%	2%
		4 min	15 min	ET-1	0%	2%
		90 sec	---	RT-6	---	75%
		---	---	RT-6	---	0%

REMOTE/FIELD FUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		3 sec	---	ET-3	---	TAGGED
		---	---	RT-5	---	OPEN
		---	60 sec	RT-6	0%	100%
		120 sec	---	RT-6	---	ON

I/O OVERRIDE SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	ON

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments

Crew assumes the watch at step 5.3.28 of IO.ZZ-0003, and continues plant startup per procedure	• RO/PO Ensure HPCI High Level Trip Reset
	• RO monitors RPV parameters to ensure proper RX startup
	• CRS directs PO to begin pre-warming of SJAE IAW HC.OP-SO.CG-0001, Condenser Air Removal Operation
	• At ~ 500 psig
	  o PO stops raising Throttle pressure and allows at least 1 Turbine Bypass Valve to OPEN
	  o CRS starting of the 2nd Secondary Condensate Pump IAW procedure
		Rx Pressure should be allowed to continue increasing to 500 psig

<u>CRS Flow controller fails downscale in AUTO</u>: After the PO stops raising Throttle pressure and allows at least 1 Turbine Bypass valve to OPEN, <u>OR</u> at the discretion of the Lead Examiner, **TRIGGER RT-1**.	• RO recognizes: ⇒ Failure of CRD flow controller and reports failure to CRS	
	• CRS directs RO to place CRD flow controller in MANUAL and attempt to open the flow control valve • RO places CRD flow controller in MANUAL ⇒ RO Depresses OPEN on CRD Flow Controller to establish normal CRD parameters	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• RO/PO contact personnel outside of Control Room to investigate failure of the CRD Flow Control Valve in AUTO.	
<u>Loss of "B" MG Set</u> Once the crew has returned CRD parameters to normal OR At the discretion of the Lead Examiner **TRIGGER – RT2**	• Crew recognizes trip of "B" MG Set	
	⇒ RO responds to Annunciators	
	⇒ RO recognizes that NO actual scram condition exists and DOES NOT scram the reactor	
	⇒ CRS enters AB-IC-0003	
	⇒ Determines Normal CANNOT be restored	
	⇒ Directs RO to Transfer power to Alternate power supply	
	• RO verifies Alternate Power is available	
	⇒ RO Transfers Power to Alternate Power Supply by Positioning the RPS MG SET TRANSFER SWITCH to the Alternate Position.	Improper operation of this will cause a scram
	• RO Resets the ½ Scram by: o Turning the key for the Affected RPS channel to the RESET position o Turning the key back to the NORMAL position o Verify the scram is reset	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• BOP verifies MSIV's are still OPEN	
	⇒ CRS directs that the tripped NSSS logic be reset	
	o RO presses the NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM TRIP LOGIC B RESET Pb.	
	o RO Verifies MSIV TRIP LOGIC TRIPPED light goes off	
	⇒ CRS Directs restoration of RWCU system	
	• RO restores RWCU system IAW applicable SOP.	
<u>Control Rod 22-35 partially scrams</u>	• RO references ARP for C6-E3, Rod Drift	
Once RWCU system has been returned to service OR At the discretion of the Lead Examiner **TRIGGER – RT3**	• RO recognizes that Rod 22-35 has inserted to position 12	
	• CRS enters AB.IC-0001, Control Rod	
	• CRS determines that Reactor Pressure < 900 psig AND Charging Water Header pressure < 940 psig AND Control Rod Scram Accumulator 22-35 is INOPERABLE and that a Manual Scram is required	** Have CRS address tech specs required for this malfunction after the scenario is over.
	⇒ CRS directs the RO to LOCK Mode Switch in SHUTDOWN	
CT-1	• RO LOCKS the Mode Switch in the SHUTDOWN Position	
	• CRS enters AB-0000	
	• RO verifies all Control Rods fully inserted	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	⇒ CRS Enter EO-101 if RPV level drops below +12.5"	
	• RO Inserts SRMs and IRMs	
	⇒ BOP verifies H2 Injection system tripped	
	• CRS directs tripping the Main turbine AND verifying Generator Lockout (MA) at 0 Mwe	
	• BOP trips the Main turbine	
	• BOP Ensures Generator Lockout	
	• CRS directs RO to maintain RPV level between +12.5" AND +54" using feedwater	
	• RO Verifies feedwater aligns to startup level control	
	• CRS directs BOP to maintain pressure within a band	
	⇒ BOP verifies EHC is controlling RPV pressure below 1037 psig	
	• BOP maintains Condenser vacuum using SJAE's	
	• BOP controls plant cooldown/ depressurization using Main Turbine Bypass valves	
RCIC Steam Leak Once the crew has stabilized the plant after the Scram OR At the discretion of the Lead Examiner **TRIGGER – RT3**	• BOP responds to annunciator D3-A2, RCIC/RHR B Area Leak Temp Hi	
	• BOP dispatches an NEO to investigate High temperature	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	BOP determines RCIC has NOT Tripped and • **PERFORMS** the following in quick succession: o **DETERMINES** the Channel that initiated this alarm from the Digital Points OR NUMAC Monitor 10C621-Z5 (1SKXR-11502) OR 10C640-Z7 (1SKXR-11503). • • **BOP CHECKs** Page 13 and Page 16 of This attachment for Channel(s) in respective monitor which initiated alarm. • **CHECK** area cooling • **MAXIMIZE** area cooling * IF continued operation is not required, **SHUT DOWN** RCIC. IF the turbine is injecting water into Reactor Vessel AND IF it is desirable to continue the operation, **PLACE** CHANNEL B(D) ISOLATION BYPASS SWITCH in BYPASS (Local Panel P621 (P640)). • CRS refers to HC.OP-EO.ZZ-0103(Q) Reactor Building Control ▪ CRS directs RO/BOP to monitor and Control Reactor Building Temps	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	⇒ BOP reports RCIC pump room temperature > Column 1 – Max Normal Op Temp	
	• CRS determines FRVS is NOT in service	
	• CRS directs BOP to verify proper operation of RBVS and Emergency Area Cooling System	
	⇒ BOP determines RBVS and Emergency Area Cooling systems are operating properly	
	• CRS directs BOP to start all available RBVS fans	
	⇒ BOP starts all available RBVS fans	
5 minutes after dispatched, NEO reports steam leak inside RCIC room	• CRS directs isolation of RCIC	
	⇒ RO trips RCIC and attempts to shut RCIC isolation valves	
	• RO determines that RCIC tripped, however, RCIC isolation valves failed to close	If asked NEO reports steam is still leaking in RCIC room.
	⇒ RO requests assistance from NEO/ WCC to close RCIC isolation valve	
	• BOP reports RCIC pump room temperature > Max Safe Op Temp	
	• CRS determines RCS is discharging into the Reactor building	
	• CRS directs a Recirc Runback and a Reactor Scram	Note – Reactor is already scrammed, so CRS may not direct this action
	• CRS enter EO-101 concurrently with this procedure	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• BOP reports 2nd area has exceeded it's Max Safe Op Limit	
	• CRS determines Emergency Depressurization is required and enters EO-0202	
	• CRS determines the following: o Reactor is shutdown from all conditions without boron o DW pressure is < 1.68 psig o Supp Pool level > 0"	
	⇒ CRS orders 5 ADS valves to be Opened	
CT-2	⇒ BOP Places all 5 ADS valve hand switches to OPEN	
Note - Failure of "E" ADS to open was input as an initial condition	• BOP recognizes PSV-F013E failed to OPEN	
	• CRS directs BOP to open non-ADS SRVs until a total of 5 SRVs are open	
CT-3	• BOP opens an additional SRV until a total of 5 SRVs are open	
<u>Termination Requirement:</u> When 5 SRV are OPEN OR At Lead Examiner Discretion		
	• CRS determines an Alert Classification is required IAW ECG Section 3.2.2.b (Valid High Drywell Pressure).	** Need to check on ECG classification **
	• Also have the CRS address the Tech Specs for the failed rod.	

VI. SCENARIO REFERENCES:

A. NC.TQ-DG.ZZ-0002 Conduct of Simulator Training.

B. NUREG 1021 Examiner Standards

C. JTA Listing

D. Probabilistic Risk Assessment

E. Technical Specifications

F. Emergency Plan (ECG)

G. Alarm Response Procedures (Various)

H. SH.OP-AS.ZZ-0001 Operations Standards

I. SH.OP-AP.ZZ-0101 Post Transient Response Requirements

J. SH.OP-AP.ZZ-0108 Operability Assessment and Equipment Control Program

K. HC.OP-IO.ZZ-0003 Startup from Cold Shutdown to Rated Power

L. HC.OP-AB.IC-0003 REACTOR PROTECTION SYSTEM

M. HC.OP-AB.IC-0001 Control Rod

N. HC.OP-AB.ZZ-000 Reactor Scram

O. HC.OP-AB.ZZ-0001 Transient Plant Conditions

P. HC.OP-EO.ZZ-0101 RPV Control

Q. HC.OP-EO.ZZ-0101A ATWS-RPV Control

R. HC.OP-EO.ZZ-0102 Primary Containment Control

S. HC.OP-EO.ZZ-0202 Emergency RPV Depressurization

T. HC.RE-IO.ZZ-0001 Core Operations Guidelines

VII. NRC CRITICAL TASK RATIONAL

NRC-001 / 00

1.

* ***Recognize that Reactor Pressure < 900 psig AND Charging Water Header pressure < 940 psig AND Control Rod Scram Accumulator 22-35 is INOPERABLE and Manually Scram within two minutes***

K/A 201001 Control Rod Drive Hydraulic System
A2.04 Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: †Scram conditions(CFR: 41.5 / 45.6) RO 3.8/ SRO 3.9

Control Rod Drive system is malfunctioning at a low reactor pressure. The reactor must be scrammed immediately to insure that all control rods are successfully inserted prior to pressure dropping below the point where the rods would insert. Two minutes is deemed adequate time to recognize the condition and implement the Immediate Operator Actions of AB.IC.0001.

2.

* ***Crew actuates five SRVS within two minutes of RCIC room temperature exceeding 250 degrees by Control Room indication (SPDS/CRIDS).***

K/A 295032 High Secondary Containment Area Temperature
EK3 Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE
EK3.01 Emergency/normal depressurization RO 3.5 SRO 3.8
EA2 Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE
EA2.01 Area temperature RO 3.8 SRO 3.8

The steam leak in the HPCI room is now affecting a second area. The reactor must be depressurized to place it in it's lowest energy state due to the potential for multiple inoperable safety systems, to reduce the driving head for the leak, and to reject decay heat to the suppression pool rather than the Reactor Building. The term "Crew actuates five SRVs" takes into account the F013D failure, which is already inserted. Two minutes is deemed adequate time to recognize the condition and implement EOP-202 and AB.ZZ-0001 Att. 13.

3.

* ***<u>WHEN</u> the PSV-F013D SRV fails to open,***
<u>THEN</u> before RPV pressure drops below 50 psig, the Crew ensures a fifth SRV is opened to achieve five open SRVs.

K/A 239002 Relief/Safety Valves
A4 Ability to manually operate and/or monitor in the control room:
A4.01 SRV's RO 4.4 SRO 4.4

The Minimum Number of SRVs required for Emergency Depressurization (MNSRED) is five. The MNSRED is utilized to assure the RPV will depressurize and remain depressurized when Emergency Depressurization is required. When the PSV-F013D fails to open, the Crew needs to open an additional SRV to achieve MNSRED. This is directed by both EOP-202 and AB.ZZ-0001. SRV's are designed to open with a minimum differential pressure of 50 psid between the reactor vessel and the suppression chamber. Below this d/p, they may not open. If the Crew does not attempt to open the fifth SRV before this minimum d/p is lost, they cannot validate it's operation. This would prevent them from detecting the failure and pursuing the use of the Alternate Depressurization Systems in EOP-202.

NRC-001 / 00

HOPE CREEK NRC - PRA RELATIONSHIPS EVALUATION FORM

EVENTS LEADING TO CORE DAMAGE

Y/N	EVENT	Y/N	EVENT
	TRANSIENTS:		SPECIAL INITIATORS:
	Turbine Trip		Loss of SSW
Y	Loss of Feedwater		Loss of SACS
	MSIV Closure		Loss of RACS
	Loss of Condenser Vacuum		Loss of Instrument Air
	Inadvertent Open SRV		
	Loss Of Offsite Power	Y	ATWS
	Station Black Out	Y	LOCA

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
	HPCI		Class 1E 120VAC Bus – A Train
	RCIC		Class 1E 120VAC Bus – D Train
	One SRV		EDG A
	One SSW Pump / Loop		EDG B
	Circulating Water System – 4 pumps		TACS

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
Y	Manual RPV Emergency Depressurization when required
	Manual RPV Depressurization during ATWS
Y	Initiation of RHR for Decay Heat Removal
	Initiation of Containment Venting
	Restore Offsite power within 45 minutes
	SACS / SSW restoration after total loss of both systems
	Avoiding Loss of Feedwater during transient
	Recovery of the Main Condenser

Complete this evaluation form for each Exam.

VIII. TURNOVER SHEET:

Rx Power: 100%
MWe: 1136 (May vary slightly):
Work Week: Any
Risk Color: Green
SMD: None
River Temp: 65

Activities Completed Last Shift:
None

Major Activities Next 12 Hours:
None

Protected Equipment:
None

Tagged Equipment:
None

NRC EXAMINATION SCENARIO GUIDE REVIEW/VALIDATION

Note: This form is used as guidance for an examination team to conduct a review for the proposed exam scenario(s). Attach a separate copy of this form to each scenario reviewed.

SELF-
CHECK NRC- 001 REVIEWER: ______________

______ 1. The scenario has clearly stated objectives in the scenario.

______ 2. The initial conditions are realistic, equipment and/or Instrumentation may be out of service, but it does not cue crew into expected events.

______ 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

______ 4. The use of non-mechanistic failures (e.g. pipe break) should be limited to one or a credible preceding event has occurred.

______ 5. The events are valid with regard to physics and thermodynamics.

______ 6. Sequencing/timing of events is reasonable (e.g. the crew has time to respond to the malfunctions in an appropriate time frame and implements procedures and/or corrective actions).

______ 7. Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.

______ 8. If time compression techniques are used, scenario summary clearly so indicates.

______ 9. The simulator modeling is not altered.

______ 10. All crew competencies can be evaluated.

______ 11. Appropriate reference materials are available (SOERs, LERs, etc.)

______ 12. If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.

______ 13. Proper critical task methodology used IAW NRC procedures.

NRC Examination Validation:

Rev.	Date	Comments

Note: The following criteria list scenario traits that are numerical in nature. A second set of numbers indicates a range to be met for a set of two scenarios. Therefore, to complete this part of the review, the set of scenarios must be available. The section below should be completed once per scenario set.

NRC: 001 **NRC:** ______

SELF-CHECK

______ 1. Total malfunctions inserted: 4-8/10-14

______ 2. Malfunctions that occur after EOP entry: 1-4/3-6

______ 3. Abnormal Events: 1-2/2-3

______ 4. Major Transients: 1-2/2-3

______ 5. EOPs used beyond primary scram response EOP: 1-3/3-5

______ 6. EOP Contingency Procedures used: 0-3/1-3

______ 7. Approximate scenario run time: 45-60 minutes (one scenario may approach 90 minutes)

______ 8. EOP run time: 40-70% of scenario run time

______ 9. Crew Critical Tasks: 2-5/5-8

______ 10. Technical Specifications are exercised during the test

______ 11. Events used in the two scenarios are not repeated

______ 12. The scenario sets for the exam week do not contain duplicate scenarios

Comments:

HOPE CREEK ELECTRIC GENERATING STATION
NRC INITIAL LICENSED EXAMINATION SCENARIO 2
NOVEMBER 28, 2005

SCENARIO TITLE: **Electrical ATWS/ SRV fails/ Small Break LOCA**

SCENARIO NUMBER: **NRC-002**

EFFECTIVE DATE:

EXPECTED DURATION: 1.0 Hours

REVISION NUMBER: **00**

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **OTHER** ____________________

REVISION SUMMARY:

New Scenario.

PREPARED BY: M. L. Brown — 9/29/05

NRC Operations Examiner — **DATE**

FACILITY REVIEWER: ____________________ ____________

Nuclear Operations Training Supervisor – Hope Creek — **DATE**

APPROVED BY: ____________________ ____________

NRC Chief Examiner — **DATE**

I. OBJECTIVE(S):

Enabling Objectives

A. The crew must demonstrate the ability to operate effectively as a team while completing a series of CREW CRITICAL TASKS, which measure the crew's ability to safely operate the plant during normal, abnormal, and emergency plant conditions.
(Crew critical tasks within this examination scenario guide are identified with an " *.")

II. MAJOR EVENTS:

A. Perform Core Spray Full Flow test
B. Power increase using Recirc Flow
C. Inadvertent HPCI initiation
D. FRVS fails to start
E. 10B130 trips
F. EHC pump trips, Electrical ATWS
G. SRV fails OPEN, Broken SRV tailpipe
H. Small Break LOCA, PSP function lost
I. RHR Spray Logic Failure

III. SCENARIO SUMMARY:

The plant is operating at 80% power, Middle Of Cycle with SLC Pump AP-208 tagged out for a motor replacement and is expected back within 48 hours. Core Spray Loop A operability PT will be performed. When the test return valve is opened, the loop flow instrumentation will fail to respond. Core Spray A should be declared Inoperable.

When power has been increased by ~10%, HPCI will inadvertently initiate. The crew will respond per AB.RPV-0001, Reactor Power, and terminate HPCI operation. A scram on high flux may occur if HPCI is not terminated. HPCI will be declared Inoperable and Tech Specs addressed.

** Talk to Steve ** Power to the Main Stack Rad Monitor will fail. The power loss results in Group 6 isolation, Secondary Containment Isolation but FRVS fails to auto start. The crew must manually start FRVS to maintain Secondary Containment integrity.

480VAC Unit Substation 10B130 will trip. This results in loss of power to the running Stator Cooling pump. The standby pump fails to auto start and must be manually started to prevent a turbine trip. The Unit Substation loss also results in a loss of Recirc Pump 2B due to oil pump B2 tripping and oil Pump B3 fails to start automatically or manually. Recirc MG Set 2B oil pressure drops below the trip setpoint but fails to trip. The MG set must be manually tripped. This places the plant in Region B - Immediate Exit region of the power to flow map. Recirculation flow must be increased or control rods must be inserted to exit Region B.

The EHC pressure regulator will fail resulting in opening of Turbine Control and Bypass valves to the Max Combined Flow limit (110%). Steam line pressure drops to the MSIV isolation setpoint and the MSIVs close. The reactor fails to auto scram and manual scram also will fail. RRCS will fail to auto initiate on high RPV pressure. The rods can be inserted by manually initiating RRCS.

RPV pressure will spike high due to the MSIV closure and RPS failure. SRVs will lift on high pressure. SRV F will fail to reclose when rods are inserted and RPV pressure lowers. The tail pipe on SRV F will rupture in the suppression chamber airspace shortly after the valve sticks open resulting in rapidly rising containment pressure and temperature.

Feed flow is lost to the RPV due to MSIV closure. HPCI may be manually started to restore RPV level but will fail shortly after being started.

If containment spray is attempted the B RHR Spray logic will fail and F016A will not OPEN. Suppression chamber pressure will rise above the safe value for Pressure Suppression Pressure requiring emergency depressurization. Low pressure ECCS and Condensate must be operated during depressurization to prevent uncontrolled injection.

When the reactor has been depressurized, the containment spray will be repaired and can be placed in service then the scenario can be terminated requiring the BOP to open another SRV. Scenario will end after 5 SRVs have been opened.

IV. INITIAL CONDITIONS:

I.C.

Initial	
______	**INITIALIZE** the simulator to 80% power, MOL
______	
______	

PREP FOR TRAINING (i.e., RM11 set points, procedures, bezel covers):

Initial	Description
______	**COMPLETE** Attachment 2 "Simulator Ready-for-Training/Examination Checklist" of NC.TQ-DG.ZZ-0002(Z).

EVENT TRIGGERS:

Initial	ET #	Description	
	1	EVENT ACTION: COMMAND: PURPOSE:	
	2	EVENT ACTION: COMMAND: PURPOSE:	
	3	EVENT ACTION: COMMAND: PURPOSE:	

MALFUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	NONE	---	---
		---	---	RT-1	---	100%
		---	---	RT-2	---	---
		---	---	RT-3	---	---
		4 min	15 min	ET-1	0%	2%
		4 min	15 min	ET-1	0%	2%
		90 sec	---	RT-6	---	75%
		---	---	RT-6	---	0%

REMOTE/FIELD FUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		---	---	NONE	---	INSTALL
		3 sec	---	ET-3	---	TAGGED
		---	---	RT-5	---	OPEN
		---	60 sec	RT-6	0%	100%
		120 sec	---	RT-6	---	ON

I/O OVERRIDE SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	ON

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
Crew assumes the watch and starts performing HC.OP-ST.BE-0002 **TRIGGER RT-1**	• BOP observes proper Core Spray pump A suction pressure	** Need to get a copy of this procedure from Archie **
	• BOP Ensures pump suction valve (HV-F001A) is Open o BOP Sends an NEO to pump to check pump out prior to start	
	⇒ BOP Starts "A" Core Spray Pump while monitoring pump discharge pressure and confirms discharge pressure rises to > 300 psig in less than or equal to 5.0 seconds • BOP Records time Core Spray pump was started • BOP ensures the following: • Core Spray Division I Room Cooler fan has started * Service Water Outlet valve is Open (NEO to report) ⇒ BOP – Throttles open Core Spray Full Flow Test Byp Valve, HV-F015A to obtain ≥ 4625 gpm flow. • BOP Determines that Core Flow indicator E21-FI-R601A fails to indicate actual flow and reports instrument malfunction • CRS directs Core Spray A to be shutdown and returned to standby lineup ⇒ CRS directs I&C to investigate ⇒ CRS refers to Tech Spec 3.5.1 and determines Action A (7 day) applies	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	⇒ BOP – shuts down Core Spray pump A and returns Core Spray to standby alignment ⇒ Close E21-F015A ⇒ Close E21-F031A	
Raise Power using Recirc Flow Once Core Spray has been returned to a standby alignment and Tech Spec call has been made OR At the discretion of the Lead Examiner **Have Load Dispatcher contact crew to raise power**	⇒ CRS – directs RO/BOP to raise power to 100% using IOP-0006.	
	• RO monitors plant for proper operation • RO refers to HC.OP-SO.BB-0002 regarding MG set critical vibration and flow instability points	
	⇒ RO – raises reactor power by increasing Recirc Flow per IOP-0006 at a rate not to exceed 1%/minute • RO slowly turns the Recirc pump Master Speed Control potentiometer in the clockwise direction.	** Ask Archie if Hope Creek typically operates with the Master Speed controller or Not **
	• RO monitors the following for proper operation • Recirc speed increases • Recirc loop flow increases • Reactor power increases	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• BOP verifies #4 STEAM LEAD DRAIN (HV-1018A) is CLOSED when the #4 CONTROL VALVE indicates off it's open seat	
INADVERTENT HPCI INITIATION Once power has been raised by ~10% OR At the discretion of the Lead Examiner **TRIGGER RT-2**	⇒ RO – diagnoses and reports inadvertent HPCI initiation	** Ask Archie if RO or CRS should direct entry into AB **
	o CRS directs entry into AB.RPV-0001 ⇒ RO verifies Reactor level > -38" ⇒ Drywell pressure < 1.68# • RO presses and holds the HPCI TURB TRIP PB • RO observes the following close • FD-FV-4880 • FD-FV-4879 • RO adjusts FIC-R600 HPCI Flow controller to 0 gpm • RO place FIC-R600 in MANUAL	
	RO - **PRESS** FIC-R600 "DECREASE" Pb for approximately 7 seconds. RO **RELEASE** the HPCI TURB TRIP PB. RO **VERIFY** the FD-FV-4879 remains shut. • BOP reduces reactor power with Reactor recirculation flow as necessary to prevent a reactor scram ⇒ CRS contacts I&C to investigate HPCI failure	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• CRS refers to Tech Spec 3.5.1. Determines Action D applies (Verify RCIC OPERABLE and restore HPCI to Operable within 14 days)	
	⇒ CRS refers to ECG and determine reportability requirements (8 hours for loss of single train)	** Check on reportability requirements **
<u>Power Loss to Main Stack Rad Monitor</u> When CRS has determined reportability requirements OR At Lead Examiner discretion **TRIGGER RT-3**	• BOP recognizes and reports loss of power to stack rad monitor	** Talk to Steve and see if he wants to do this malfunction
	• BOP refers to ARP's and verifies auto actions	
	• BOP/CRS determines FRVS did not start as required	
	• CRS directs starting of FRVS	
	• BOP starts FRVS	
	• Place FRVS A and B control switches *to ON and verifies* negative pressure restored by observing pressure indication and alarm clearing	
	• BOP determines Group 6 has isolated	
	⇒ BOP determines that Secondary Containment has isolated	
	• BOP dispatches NEO to investigate power loss	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• CRS directs I&C to investigate and determine if stack rad monitor can be transferred to alternate power supply • CRS refers to Tech Spec 3.3.6.1 (PCIS Instrumentation) determine Function 2c is inoperable, determine actions A & B apply (single channel), then C and F after 1 hour • CRS - Refer to TRM 3.4 (post accident monitoring), determine Function 5 is inoperable and Condition A applies • CRS - Refer to ODCM 7.3.2 (gaseous effluent monitoring), determine function 1 is inoperable, Conditions A & B apply (grab samples) and notify Chemistry	
<u>480V Unit Substation 10B130 trips</u> Once Tech Specs and ODCM have been addressed OR At the discretion of the Lead Examiner **TRIGGER RT-4**	Crew responds to loss of 10B130	** Check with Archie to ensure pump should be tripped ** Check with Steve and either have 1B2P120 either fail to start or have it tagged out for maintenance
	⇒ RO diagnose failure of Recirc Pump B to trip and manually trip ⇒ Place B Recirc pump MG-Set supply breaker to OFF • CRS direct entry to AB.RPV-0003 ⇒ RO – inserts rods to clear APRM Upscale Alarms	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	RO - **ENSURE** that the Recirc MG Drive Motor Breaker has TRIPPED for the tripped Pump.	
	RO - **CLOSE** HV-F031A(B) for approximately 5 minutes, THEN **RE-OPEN** HV-F031A(B). **[CD-976B]**	
	RO - **IMPLEMENT** the following: • DL.ZZ-0026 Att. 3n (as required) • DL.ZZ-0026 Att. 3v	
	CRS - **DIRECT** the Reactor Engineer to develop a Rod Sequence to achieve an 80% Rod Line.	
	CRS - **IMPLEMENT** IO-6 Requirements for Single Loop operations.	
	CRS determines region of operation on power/flow map	
	CRS directs actions to exit Region B	
	• RO either Raises Recirc flow with Recirc pump A or inserts control rods to exit Region B	
	• CRS refers to Tech Spec 3.4.1 and COLR for SLO, determine APLHGR limit and APRM setpoints must be modified within 6 hours (4 hours per IOP-6)	
	• CRS contacts I&C to determine cause of failure and to adjust setpoints as required	
	• CRS refers to IOP-6 and determines all appropriate actions have been taken in accordance with Section 5.3	** See if anything else needs to be done for single loop operation

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
		** See if any other actions need to be taken for loss of 10B130 and where guidance may be obtained
<u>EHC Failure/ Electrical ATWS/ "F" SRV fails to close</u> After CRS has notified I&C to adjust setpoints OR At the discretion of the Lead Examiner **TRIGGER RT-5**	⇒ Crew diagnoses failure of EHC pressure control	• Malfunctions to be inserted Pressure regulator fails high (or #4 control valve fails open, something to drag pressure down) Auto Scram defeat Manual Scram defeat "F" SRV fails to close after opening
	• RO – recognizes MSIV closure and failure to auto scram • CRS directs manual scram and entry into EO.ZZ-0101 • RO – manually scrams the reactor by depressing manual scram pushbuttons • RO – recognizes the failure of the manual scram and places Mode Switch to SHUTDOWN	
	• CRS – directs RRCS to be initiated if not already completed by RO • CRS enters EO.ZZ-0101A if rods are not yet inserted	
CT-1	• RO manually initiates RRCS • Place RRCS keylock in Trip • Place RRCS CS in Trip	
	• RO reports when all rods are inserted	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• BOP recognizes MSIV closure and ensures pressure is controlled by the SRVs as required. • CRS exits EO.ZZ-0101A after control rods are inserted, returns to EO.ZZ-0101	
<u>"F" SRV Fails to Close/HPCI failure</u> Once Rods are inserted OR At Lead Examiner Discretion **TRIGGER RT-6**	• RO/BOP diagnoses the failure of "F" SRV to close	** Talk to Archie with plant tripped and 1 SRV open is HPCI needed or will plant depressurize down to CS/RHR entry conditions • Note "F" SRV failure is inserted as an Initial condition
	• RO attempts to close "F" SRV • CRS directs actions of AB.RPV-0006 for SRV failure while continuing in EO.ZZ-0101 • CRS performs actions for EO.ZZ-0102 as appropriate	
	• IAW AB>RPV-0006, RO reduces Recirc pump speed to minimum	** See if RO has to do anything for this
	• RO cycles "F" SRV control switch several times to attempt to close the SRV • BOP ensures MSIV's and HV-F016 and HV-F019 are closed to attempt to control cooldown • RO starts suppression pool cooling IAW AB-0001 • BOP breaks condenser vacuum as follows • Verify Main Turbine < 1200 rpm • OPEN HV-1972 A/B/C	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• RO attempts to start HPCI to control RPV level	** Talk to Archie/ Steve and see what is best method to fail HPCI
	• RO – diagnoses and reports failure of HPCI to operate	
<u>**"F" SRV Tailpipe Rupture, CNMT Spray Failure/ Small Break LOCA**</u> Once RO has initiated Suppression Pool Cooling OR At Lead Examiner Discretion **TRIGGER RT- 7**	• RO/BOP – diagnose and report rapidly rising containment pressures	** Talk to Archie – Goal it push operators to ED ** Talk to Steve – not sure why we are putting LOCA in here. See what he says. How do you expect operators to react HV-F027A, B failed closed *Maybe override PB's don't work
	• RO/BOP determine leak is in suppression chamber based on higher suppression chamber pressure and/or vacuum breaker operation	
	• CRS directs initiation of suppression chamber spray per EO.ZZ-0102	
	• CRS directs initiation of drywell spray per EO.ZZ-0102	
	• RO/BOP Attempt to initiate spray per SO.BC-0001	
	• RO – Press BC-HV-F027B RHR LOOP B SUPP CHAMBER SPRAY HDR ISLN MOV AUTO CL OVRD PB	
	• RO – Attempts to OPEN HV-F027B	F027B won't open

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• RO – diagnoses Drywell and Suppression pool spray can't be initiated	
	• CRS – When PSP limits exceeded, determine Emergency Depressurization is required.	
	• CRS – Directs entry to EO.ZZ-0202 when appropriate	
	• CRS/RO determines Reactor is shutdown under all conditions without boron	
	• CRS/RO determines Drywell pressure > 1.68 psig	
	• CRS directs RO to prevent injection from Core Spray and LPCI pumps not required to assure adequate core cooling	
	• RO overrides Core Spray and RHR pumps not required for core cooling to Off	
	• CRS/BOP determines Suppression Pool level > 0"	
	• CRS directs RO/BOP to OPEN 5 ADS valve and Defeat PCIG *isolation interlocks if necessary*	** When would it be necessary to defeat PCIG interlocks – What are we talking about here.
CT-2	• RO opens 5 ADS valves	
	• BOP operate Condensate system to prevent uncontrolled injection	
	•	** Talk to Steve about stopping point
	•	

VI. SCENARIO REFERENCES:

A. NC.TQ-DG.ZZ-0002 Conduct of Simulator Training.

B. NUREG 1021 Examiner Standards

C. JTA Listing

D. Probabilistic Risk Assessment

E. Technical Specifications

F. Emergency Plan (ECG)

G. Alarm Response Procedures (Various)

H. SH.OP-AS.ZZ-0001 Operations Standards

I. SH.OP-AP.ZZ-0101 Post Transient Response Requirements

J. SH.OP-AP.ZZ-0108 Operability Assessment and Equipment Control Program

K. HC.OP-IO.ZZ-0003 Startup from Cold Shutdown to Rated Power

L. HC.OP-AB.IC-0003 REACTOR PROTECTION SYSTEM

M. HC.OP-AB.IC-0001 Control Rod

N. HC.OP-AB.ZZ-000 Reactor Scram

O. HC.OP-AB.RPV-0001 Reactor Power

P. HC.OP-EO.ZZ-0101 RPV Control

Q. HC.OP-EO.ZZ-0101A ATWS-RPV Control

R. HC.OP-EO.ZZ-0102 Primary Containment Control

S. HC.OP-EO.ZZ-0202 Emergency RPV Depressurization

T. HC.RE-IO.ZZ-0001 Core Operations Guidelines

U. HC.OP-IO.ZZ-0006, POWER CHANGES DURING OPERATION

V.

VII. NRC CRITICAL TASK RATIONAL

NRC-002 / 00

1.

* ***Before Reactor Water Level reaches LVL 1, the Crew manually actuates RPS and/or ARI to shutdown the reactor.***

K/A 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EA1. Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:
EA1.01 Reactor Protection System RO 4.6 SRO 4.6
EA1.03 ARI/RPT/ATWS RO 4.1 SRO 4.1

RPS has failed to scram the reactor both manually, and automatically. The RPV LVL 3 scram setpoint was chosen to ensure there is adequate protection for the fuel during transient analyses associated with coolant inventory decrease events. With no feedwater being supplied to the vessel and the reactor at power, water level will rapidly lower until the reactor is shutdown and steaming is reduced to decay heat levels. Additionally, ARI is failed and will not automatically scram the reactor at –38". Operator action is required to shutdown the reactor. The need to manually initiate ARI or RPS by LVL 1 was chosen because it represents an acceptable level of performance considering the rate of RPV water level reduction in this scenario and the time required to implement the scram hard card. Also, if the plant is not scrammed by LVL 1, the subsequent shrink will reduce level to below TAF.

2.

* ***Crew actuates five SRVs before Suppression Chamber pressure exceeds 33 psig.***

K/A 295024 High Drywell Pressure
EA2 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:
EA2.04 Suppression chamber pressure RO 3.9 SRO 3.9
K/A 223001 Primary Containment Systems and Auxiliaries
A2. Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions of operations:
A2.02 Steam bypass of the suppressions pool RO 3.9 SRO 4.1

If suppression chamber pressure cannot be maintained below the pressure suppression pressure, EOPs direct actions to emergency depressurize the reactor. A LOCA condition while in the action required region of the Pressure Suppression Pressure curve, could cause design containment limits to be exceeded. Based upon the rate of pressure increase in this scenario, the upper limit of 33 psig is established to give the operator time to evaluate conditions and direct emergency depressurization actions.

NRC-002 / 00

HOPE CREEK NRC - PRA RELATIONSHIPS EVALUATION FORM

EVENTS LEADING TO CORE DAMAGE

Y/N	EVENT	Y/N	EVENT
	TRANSIENTS:		SPECIAL INITIATORS:
	Turbine Trip		Loss of SSW
Y	Loss of Feedwater		Loss of SACS
	MSIV Closure		Loss of RACS
	Loss of Condenser Vacuum		Loss of Instrument Air
	Inadvertent Open SRV		
	Loss Of Offsite Power	Y	ATWS
	Station Black Out	Y	LOCA

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
	HPCI		Class 1E 120VAC Bus – A Train
	RCIC		Class 1E 120VAC Bus – D Train
	One SRV		EDG A
	One SSW Pump / Loop		EDG B
	Circulating Water System – 4 pumps		TACS

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
Y	Manual RPV Emergency Depressurization when required
	Manual RPV Depressurization during ATWS
Y	Initiation of RHR for Decay Heat Removal
	Initiation of Containment Venting
	Restore Offsite power within 45 minutes
	SACS / SSW restoration after total loss of both systems
	Avoiding Loss of Feedwater during transient
	Recovery of the Main Condenser

Complete this evaluation form for each Examination.

VIII. TURNOVER SHEET:

Rx Power: 80%
MWe: (May vary slightly):
Work Week: Any
Risk Color: Green
SMD: None
River Temp: 65

Activities Completed Last Shift:
Power lowered to 80% and
Control Rod Sequence Exchange performed

Major Activities Next 12 Hours:
Maintain power at 80% until contacted by the Load Dispatcher, then return to 100% power

Complete HC.OP-ST.BE-0002, Core Spray Pump Loop A Full Flow Test. Currently in progress and completed up to step 5.23 (pump testing).

Protected Equipment:
None

Tagged Equipment:
SLC Pump AP-208 is tagged out for pump rebuild and is expected back within 48 hours
OPRM System is INOPERABLE due to an existing 10CFR21 issue. The OPRM System is still functional but is considered INOPERABLE per Technical Specifications.
No other equipment is Out of Service

NRC EXAMINATION SCENARIO GUIDE REVIEW/VALIDATION

Note: This form is used as guidance for an examination team to conduct a review for the proposed exam scenario(s). Attach a separate copy of this form to each scenario reviewed.

SELF-
CHECK NRC- 002 REVIEWER: ______________________

______ 1. The scenario has clearly stated objectives in the scenario.

______ 2. The initial conditions are realistic, equipment and/or Instrumentation may be out of service, but it does not cue crew into expected events.

______ 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

______ 4. The use of non-mechanistic failures (e.g. pipe break) should be limited to one or a credible preceding event has occurred.

______ 5. The events are valid with regard to physics and thermodynamics.

______ 6. Sequencing/timing of events is reasonable (e.g. the crew has time to respond to the malfunctions in an appropriate time frame and implements procedures and/or corrective actions).

______ 7. Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.

______ 8. If time compression techniques are used, scenario summary clearly so indicates.

______ 9. The simulator modeling is not altered.

______ 10. All crew competencies can be evaluated.

______ 11. Appropriate reference materials are available (SOERs, LERs, etc.)

______ 12. If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.

______ 13. Proper critical task methodology used IAW NRC procedures.

NRC Examination Validation:

Rev.	Date	Comments

Note: The following criteria list scenario traits that are numerical in nature. A second set of numbers indicates a range to be met for a set of two scenarios. Therefore, to complete this part of the review, the set of scenarios must be available. The section below should be completed once per scenario set.

NRC: 002 **NRC:** ____

SELF-CHECK

____ 1. Total malfunctions inserted: 4-8/10-14

____ 2. Malfunctions that occur after EOP entry: 1-4/3-6

____ 3. Abnormal Events: 1-2/2-3

____ 4. Major Transients: 1-2/2-3

____ 5. EOPs used beyond primary scram response EOP: 1-3/3-5

____ 6. EOP Contingency Procedures used: 0-3/1-3

____ 7. Approximate scenario run time: 45-60 minutes (one scenario may approach 90 minutes)

____ 8. EOP run time: 40-70% of scenario run time

____ 9. Crew Critical Tasks: 2-5/5-8

____ 10. Technical Specifications are exercised during the test

____ 11. Events used in the two scenarios are not repeated

____ 12. The scenario sets for the exam week do not contain duplicate scenarios

Comments:

HOPE CREEK ELECTRIC GENERATING STATION
NRC INITIAL LICENSED EXAMINATION SCENARIO 3
NOVEMBER 28, 2005

SCENARIO TITLE: **APRM Failure/ Recirc Pump Hi Vibs/ LOP**

SCENARIO NUMBER: **NRC-003**

EFFECTIVE DATE:

EXPECTED DURATION: 1.0 Hours

REVISION NUMBER: **00**

PROGRAM:

- [ ] **L.O. REQUAL**
- [X] **INITIAL LICENSE**
- [ ] **OTHER** ______________________

REVISION SUMMARY:

New Scenario.

PREPARED BY: M. L. Brown — 9/29/05
NRC Operations Examiner — **DATE**

FACILITY REVIEWER: ______________________ — ______________
Nuclear Operations Training Supervisor – Hope Creek — **DATE**

APPROVED BY: ______________________ — ______________
NRC Chief Examiner — **DATE**

I. OBJECTIVE(S):

Enabling Objectives

A. The crew must demonstrate the ability to operate effectively as a team while completing a series of CREW CRITICAL TASKS, which measure the crew's ability to safely operate the plant during normal, abnormal, and emergency plant conditions.
(Crew critical tasks within this examination scenario guide are identified with an " *.")

II. MAJOR EVENTS:

A. Start 3rd RFP
B. Load increase after RFP start
C. "A" APRM Fails
D. Drywell Chiller Compressor fails
E. "B" Recirculation Pump High Vibration
F. Loss of Offsite Power
G. "A" EDG Output breaker fails to Auto close
H. Recirc Suction pipe leak

III. SCENARIO SUMMARY:

The plant is operating at 80% power, Middle Of Cycle returning to power after a mini-outage. The Operators are at Step 5.4.48 of IOP-3, preparing to start the 3rd RFP per HC.OP-SO.AE-0001, Feedwater System Operation.

After starting the 3rd RFP, the operators are to raise power to 100% by increasing recirc flow. While raising power the "A" APRM fails causing the crew to enter Abnormal Procedure HC.OP-AB.IC-0004,NEUTRON MONITORING and bypass the APRM.

After the APRM is bypassed, the "A" Drywell Chiller Compressor will fail. Drywell pressure will rise causing the operators to enter AB.CONT-0001, Drywell Pressure. When pressure rises to > 0.75 psig, Operators will vent drywell. Once drywell pressure is lowering, "B" Recirc pump vibrations will increase causing the operators to enter AB.RPV-0003, Recirculation system. Operators will reduce recirc pump speed in an attempt to clear the vibration alarm. Vibration will continue to increase and cause the operators to trip the "B" Recirc pump on high vibration. After tripping the Recirc pump the operators will have to insert rods to exit Region 2 on the Power flow map. As rods are being inserted a Loss of Offsite power occurs with the "A" EDG output breaker failing to close. In addition, HPCI and RCIC fail to auto start. The loss of offsite power will cause a scram to occur, shortly after the scram occurs a small recirc suction pipe leak occurs, the operators will be forced to either restart HPCI and feed the reactor or Emergency Depressurize. The scenario will end once the operators either stabilize level or Emergency Depressurize.

IV. INITIAL CONDITIONS:

IC

Initial	
	INITIALIZE the simulator to 80% power, MOL

PREP FOR TRAINING (i.e., RM11 set points, procedures, bezel covers):

Initial	Description
	COMPLETE Attachment 2 "Simulator Ready-for-Training/Examination Checklist" of NC.TQ-DG.ZZ-0002(Z).

EVENT TRIGGERS:

Initial	ET #	Description	
	1	EVENT ACTION: COMMAND: PURPOSE:	
	2	EVENT ACTION: COMMAND: PURPOSE:	
	3	EVENT ACTION: COMMAND: PURPOSE:	

MALFUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val

REMOTE/FIELD FUNCTION SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val

I/O OVERRIDE SUMMARY:

Initial	Description	Delay	Ramp	Trigger	Init Val	Final Val
		---	---	NONE	---	ON

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
Crew assumes the watch and starts performing HC.OP-SO.AE-0001 section 5.6.1 (Note – Feedpump should be on recirc)	• BOP OPENS HV-1769C, RFP C Discharge Stop Check valve	
	• BOP closes HV-1772C, RFPT C Steam Low Pressure supply stop valve below seat drain o BOP opens HV-1751C, RFPT C Low pressure steam isolation valve ⇒ BOP depresses the "SEL" push-button as required to select "DEMAND" on the in-service RFPT(s) whose demand will be matched • BOP Presses "SEL" push-button for the C RFPT to select "SPEED CTRLR DMND" • BOP Presses Increase or decrease buttons as necessary to equalize demand signals while Monitoring: • RFPT Discharge Pressure • RFPT DEMAND • * RFPT "FLOW"	
	⇒ BOP matches Flow and speed and transfers RFPT C Speed Control to automatic by depressing the A/M push-button and observing "A" illuminates • BOP reports to CRS that 3rd RFP has been placed in service	
<u>Power Increase</u>	• CRS directs RO/BOP to coordinate Power increase to 90% at < 1%/minute using IOP-0003	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	⇒ RO/BOP coordinate raising power	
	⇒ RO slowly increases Recirc pump speed	
	⇒ BOP monitors RFP speed to ensure proper response	
<u>"A" APRM Fails</u> After Power has been raised 5% OR At the discretion of the Lead Examiner **TRIGGER RT-1**	⇒ RO diagnoses and reports "A" APRM has failed UPSCALE	Should get a Half scram
	• CRS acknowledges report and enters HC.IO-AB.IC-0004, Neutron Monitoring	
	• RO stops all Control Rod Withdrawals	Should not be any control rod withdrawals in progress
	• RO bypasses the "A" APRM	
	• RO ensures all RPS trip conditions are clear	
	⇒ RO turns the "A" RPS Trip logic key to reset and returns it to the normal position	
	o RO verifies that RPS is reset	
	⇒ CRS refers to Tech Specs 3.3.1	Should only be an INFO only LCO – only required to have 2 OPERABLE

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
<u>TURBINE BLDG CHILLER COMPRESSOR "A" FAILS</u> Once the CRS has addressed Tech Specs OR At the discretion of the Lead Examiner **TRIGGER RT-2**	• BOP diagnoses/ observes the "A" Turbine Building Chiller trips	** Goal here is to have Drywell pressure increase to the point where the operators need to perform a drywell vent.
	• RO/BOP observe Drywell temperature/ pressure rising • CRS directs entry into AB.CONT-0001, Drywell Pressure	
	RO - **TERMINATE** Drywell Inerting.	No drywell cooling in progress
	RO **MAXIMIZE** Drywell Cooling by **ENSURING**: • All Drywell Fan Cooling Coils are Open. • All Drywell Fans are running in Fast Speed. • Turbine Bldg. Chill Water system is operating properly.	Turbine Bldg. Chilled water system is NOT operating properly
	RO - **PERFORM** the following: • Check Reactor Recirc. Pump Seals. • Check SRV Tailpipe Temperatures.	
**** Talk to Archie/Steve what is the best/ quickest way to raise Drywell pressure so crew will vent containment**		

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
<u>"B" Recirc Pump High Vibration</u> Once Drywell vent has been initiated OR At the discretion of the Lead Examiner **TRIGGER RT-3**	RO diagnoses/ observes rising "B" Recirc pump vibration	
	CRS directs entry into AB.RPV-0003, Recirculation System	
	RO <u>PRIOR</u> to reducing Recirc Pump Speed, **PERFORM** the following: **ENSURE** the following controllers are in MANUAL • SIC-R621A PUMP A SPD CONT • SIC-R621B PUMP B SPD CONT	
	RO **RECORD** affected pump speed: • B Recirc Initial Pump Speed	
	RO **MAINTAIN** the affected Pump ALERT limit [**REFER** to Table 2] clear as follows: • INTERMITTENTLY **PRESS** SIC-R621A(B) PUMP A(B) SPD CONT DECREASE push button on the affected Recirculation Pump. • **INSERT** Control Rods as required by Reactor Engineering Instructions.	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	RO IF ALERT limit cannot be maintained clear {**REFER** to Table 2] AND the affected Recirculation Pump Speed has been lowered by ≥20% (below the value logged in Step K.1.B), THEN **REMOVE** the affected Recirc Pump from service IAW HC.OP-SO.BB-0002, Single Loop Operation.	
CT-1	RO removes pump from service IAW SO.BB-0002	
	CRS - **IMPLEMENT** IO-6 Requirements for Single Loop operations.	
	CRS determines region of operation on power/flow map	
	CRS directs actions to exit Region B	
	• RO either Raises Recirc flow with Recirc pump A or inserts control rods to exit Region B	
	• CRS refers to Tech Spec 3.4.1 and COLR for SLO, determine APLHGR limit and APRM setpoints must be modified within 6 hours (4 hours per IOP-6)	
	• CRS contacts I&C to determine cause of failure and to adjust setpoints as required	
	• CRS refers to IOP-6 and determines all appropriate actions have been taken in accordance with Section 5.3	** See if anything else needs to be done for single loop operation
LOSS OF OFFSITE POWER After CRS has notified I&C to adjust setpoints OR At the discretion of the Lead Examiner **TRIGGER RT-4**	⇒ Crew diagnoses Loss of Offsite power	• Malfunctions to be inserted Loss of Offsite power "A" EDG Output breaker fails to Auto Close HPCI fails to auto start RCIC fails to auto start

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
	• CRS directs entry into HC.OP-AB-0000 and HC.OP-AB.ZZ-0135	**** Ask Archie if it is expected to have level drop < 12.5" and force entry into EO-101**
	• RO locks Mode Switch in Shutdown	
	• BOP observes failure of "A" EDG to Auto close and Closes EDG output breaker	** Check with Archie if this is the appropriate action
	• RO verifies the Scram	
	• RO inserts SRMs and IRMs AND selects IRMs on the Recorders	
	• BOP verifies H2 injection system tripped	
	• BOP Trips the Main turbine and verifies Generator lockout is 0 Mwe	
	• RO maintains level between +12.5" and 54"	
	• RO starts RCIC	
<u>RECIRC PIPING LEAK</u> Once LOP loads have sequenced on OR At the discretion of the Lead Examiner **TRIGGER RT-5**	• RO observes RPV level decreasing, and Drywell temperature/ pressure increasing	
	• CRS directs entry into EO-0101 and EO-0102 based on Drywell pressure > 1.68 psig and level < 12.5"	Crew may elect to auto start HPCI and control level, if this occurs they may not observe the failure of HPCI to auto start
CT-2	• RO starts HPCI and controls level	

V. SCENARIO GUIDE SEQUENCE AND EXPECTED RESPONSE

Event / Instructor Activity	Expected Plant/Student Response	Comments
After HPCI has been started and level is being controlled, scenario may be terminated.	•	
	• Once Scenario has been terminated have the SRO classify the event.	
	•	** Talk to Steve about stopping point
	•	

VI. SCENARIO REFERENCES:

A. NC.TQ-DG.ZZ-0002 Conduct of Simulator Training.

B. NUREG 1021 Examiner Standards

C. JTA Listing

D. Probabilistic Risk Assessment

E. Technical Specifications

F. Emergency Plan (ECG)

G. Alarm Response Procedures (Various)

H. SH.OP-AS.ZZ-0001 Operations Standards

I. SH.OP-AP.ZZ-0101 Post Transient Response Requirements

J. SH.OP-AP.ZZ-0108 Operability Assessment and Equipment Control Program

K. HC.OP-IO.ZZ-0003 Startup from Cold Shutdown to Rated Power

L. HC.OP-AB.IC-0003 REACTOR PROTECTION SYSTEM

M. HC.OP-AB.IC-0001 Control Rod

N. HC.OP-AB.ZZ-000 Reactor Scram

O. HC.OP-AB.RPV-0001 Reactor Power

P. HC.OP-EO.ZZ-0101 RPV Control

Q. HC.OP-EO.ZZ-0101A ATWS-RPV Control

R. HC.OP-EO.ZZ-0102 Primary Containment Control

S. HC.OP-EO.ZZ-0202 Emergency RPV Depressurization

T. HC.RE-IO.ZZ-0001 Core Operations Guidelines

U. HC.OP-IO.ZZ-0006, POWER CHANGES DURING OPERATION

V.

VII. NRC CRITICAL TASK RATIONAL

NRC-003 / 00

1.

* ***CREW secures "B" Reactor Recirc pump within two minutes of Vibration reaching the DANGER limit IAW guidance in AB.RPV-0003..***

K/A 202001 Recirculation System
A2 Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.17 Loss of seal cooling water RO 3.1 SRO 3.2

This action is listed as a Retainment Override in the Abnormal Procedure, a time limit of 2 minutes is deemed adequate for the operator to recognize the condition and take the appropriate action.. The basis of this action is to prevent pump damage and potential piping damage due to vibration. Damage to the pump casing is a degradation of a Reactor Coolant System boundary.

2.

* ***Before RPV water level reaches -161" and without Emergency Depressurizing, CREW manually places HPCI in service and injects with HPCI to maintain Reactor water level above -161".***

K/A 206000 High Pressure Coolant Injection System
A3 Ability to monitor the operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including:
A3.03 System lineup RO 3.9 SRO 3.8
K/A 295031 Reactor Low Water Level
EA1. Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL:
EA1.02 High Pressure Coolant Injection RO: 4.5 SRO 4.5

HPCI has failed to automatically start. HPCI is the only High Pressure injection system available with adequate capacity to maintain RPV water level. If RPV water level is allowed to drop below –161", the fuel will be uncovered and the fuel cladding will be challenged. This would escalate the event to a General Emergency. HC.OP-AB.ZZ-0001 Attachment 6 has the necessary guidance to step the operator through manually initiating HPCI in the injection mode. The rate of level drop in this scenario is very slow and provides more than adequate time to execute the guidance an restore RPV level with HPCI.

NRC-002 / 00

HOPE CREEK NRC - PRA RELATIONSHIPS EVALUATION FORM

EVENTS LEADING TO CORE DAMAGE

Y/N	EVENT	Y/N	EVENT
	TRANSIENTS:		SPECIAL INITIATORS:
	Turbine Trip		Loss of SSW
Y	Loss of Feedwater		Loss of SACS
	MSIV Closure		Loss of RACS
	Loss of Condenser Vacuum		Loss of Instrument Air
	Inadvertent Open SRV		
	Loss Of Offsite Power	Y	ATWS
	Station Black Out	Y	LOCA

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
	HPCI		Class 1E 120VAC Bus – A Train
	RCIC		Class 1E 120VAC Bus – D Train
	One SRV		EDG A
	One SSW Pump / Loop		EDG B
	Circulating Water System – 4 pumps		TACS

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
Y	Manual RPV Emergency Depressurization when required
	Manual RPV Depressurization during ATWS
Y	Initiation of RHR for Decay Heat Removal
	Initiation of Containment Venting
	Restore Offsite power within 45 minutes
	SACS / SSW restoration after total loss of both systems
	Avoiding Loss of Feedwater during transient
	Recovery of the Main Condenser

Complete this evaluation form for each Examination.

VIII. TURNOVER SHEET:

Rx Power: 80%
MWe: (May vary slightly):
Work Week: Any
Risk Color: Green
SMD: None
River Temp: 65

Activities Completed Last Shift:
Power lowered to 80% and
Control Rod Sequence Exchange performed

Major Activities Next 12 Hours:
Maintain power at 80% until contacted by the Load Dispatcher, then return to 100% power

Complete HC.OP-ST.BE-0002, Core Spray Pump Loop A Full Flow Test. Currently in progress and completed up to step 5.23 (pump testing).

Protected Equipment:
None

Tagged Equipment:
SLC Pump AP-208 is tagged out for pump rebuild and is expected back within 48 hours
OPRM System is INOPERABLE due to an existing 10CFR21 issue. The OPRM System is still functional but is considered INOPERABLE per Technical Specifications.
No other equipment is Out of Service

IX. SIMULATOR NRC REVIEW/VALIDATION CHECKLIST

NRC EXAMINATION SCENARIO GUIDE REVIEW/VALIDATION

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SELF-CHECK   NRC- 002   REVIEWER: ____________

_______ 1. The scenario has clearly stated objectives in the scenario.

_______ 2. The initial conditions are realistic, equipment and/or Instrumentation may be out of service, but it does not cue crew into expected events.

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NRC Examination Validation:

Rev.	Date	Comments

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NRC: 002 **NRC:** ____________

SELF-CHECK

______ 1. Total malfunctions inserted: 4-8/10-14

______ 2. Malfunctions that occur after EOP entry: 1-4/3-6

______ 3. Abnormal Events: 1-2/2-3

______ 4. Major Transients: 1-2/2-3

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______ 6. EOP Contingency Procedures used: 0-3/1-3

______ 7. Approximate scenario run time: 45-60 minutes (one scenario may approach 90 minutes)

______ 8. EOP run time: 40-70% of scenario run time

______ 9. Crew Critical Tasks: 2-5/5-8

______ 10. Technical Specifications are exercised during the test

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______ 12. The scenario sets for the exam week do not contain duplicate scenarios

Comments: