FPL Energy
Seabrook Station

FPL Energy Seabrook Station
P.O. Box 300
Seabrook, NH 03874
(603) 773-7000

NOV 10 2004

Docket 50-443
SBK-L-04105

United States Nuclear Regulatory Commission
Attn: Mr. Stephen Barr, Examiner/Inspector
475 Allendale Road
King of Prussia, PA 19406-1415

Seabrook Station
Submittal of Information for Operator Licensing Examinations

In a letter dated September 30, 2004, FPL Energy Seabrook, LLC received a request to furnish documentation to support the administration of Senior Reactor Operator and Reactor Operator licensing examinations scheduled for the week of January 31, 2005. The letter requested examination outlines be provided by November 15, 2004 with the written examinations, operating tests, and supporting reference materials identified in Attachment 2 of ES-201, "Initial Licensing Examination Process; Examination Security," to be furnished by December 13, 2004.

Per your request, FPL Energy Seabrook, LLC encloses the examination outlines as follows:

ES-201-2, Examination Outline Quality Checklist
ES-301-1, Administrative Topics
ES-301-2, Control Room/In-Plant Systems Outline
ES-301-5, Transient and Event Checklist
ES-D-1s, Scenario Outlines
ES-401-2, PWR Examination Outline
ES-401-3, Generic Knowledges and Abilities Outline
ES-401-4, Record of Rejected K/As
Seabrook Reference Material Disks (Index on each disk)
Outline of Sampling Methodology
Draft Exam Week Schedule

The information provided in the enclosure is confidential in nature and is not for release to the public.

Should you have any questions regarding this matter, please contact Mr. Kerry P. Wright, Operations Training Supervisor, at (603) 773-7627.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

cc (Without Enclosures):
Document Control Desk
S. J. Collins, NRC Region I Administrator
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RANDOM SAMPLING PROCESS USED TO DEVELOP 2005 NRC EXAM

As required by NUREG 1021, Rev 9 (ES-401 D.1.b), the licensee shall submit along with the NRC examination outline, a description of the systematic and random selection process used to develop the examination outline.

The most recent version (June 2004) of the W/D Associates random examination generation program received through the Westinghouse Owners Group (WOG) was used to automatically generate both the Audit and NRC Examination RO and SRO outlines. The following additional detail is provided:

- Attachment 1 & 2 of ES-401, were used as guidance for K/A elimination prior and after automatic exam outline generation.
- **Prior** to exam outline generation, the following K/As ONLY were suppressed/eliminated within the program:
    - Generic K/A statements 2.2.3 and 2.2.4 (multi-unit) – Seabrook is a single unit facility.
    - All K/As associated with Non-Nuclear Instrumentation (016) – This system is N/A to Seabrook.
    - All K/As associated with Ice Condenser (025) – This system is N/A to Seabrook.
    - All K/As associated with Containment Iodine Removal (27) – This system is included as part of the Containment Building Spray System at Seabrook.
- A separate RO and SRO outline was automatically generated using the K/A program for both the Audit and NRC examinations. Prior to accepting and saving each automatically generated outline, the verification feature of this program was run to ensure the generated outline met NRC criteria (i.e.: no K/A's < 2.5, required number of questions etc....). Once an acceptable outline was generated it was saved.
- **Following** examination outline generation, any K/A statements eliminated and resampled are documented on ES-401-4, Record of Rejected K/As that is provided as required by ES-401, Attachment 2.
- In all cases when manual resampling occurred utilizing tokens (i.e.: rejected K/A or generic statements were not relevant), the resampling occurred in the same K/A category to ensure examination balancing as automatically generated.
- It should also be noted that although ES-401, Attachment 2, paragraph 1 allows certain generic K/A's to be rejected without explanation, that justification was provided nonetheless.

Lastly, the compare function was used to identify any overlapping K/A's between the Audit and NRC RO exams and likewise the same for the SRO exams. No overlapping K/As were found between the Audit and NRC SRO outlines. Six overlapping K/A's were identified between the Audit and NRC RO outlines. When developing the exam, care will be given to ensure these questions are not similar in nature.

RANDOM SAMPLING PROCESS USED TO DEVELOP 2005 NRC EXAM

During supervisory review of the generated outlines for the licensing exam, it was noted that the point values on the SRO outline as generated by the WD Associates WOG sponsored program were different than that required by ES-401-2 Rev. 9. Specifically the Tier 1 Group 1 and Group 2 reflected the ***Draft Rev. 9*** values vice Revision 9 values. (7 to 6 and 5 to 4 respectively). Similarly, in Tier 2 Group 1 the change was 4 to 5 and Tier 2 Group 2 from 2 to 3. A station condition report was generated to track this issue (CR 04-10322). The Lead Examiner was informed of the issue as was the WOG. Within a week WD Associates provided us with a software patch to correct the sampling program. The Lead Examiner and NRC Program Office were consulted regarding an acceptable recovery plan for the SRO written examination outline. In accordance with this guidance K/A's were randomly deleted from Tier 1 and two were randomly selected and added to Tier 2. Specifically, 007 EA2.02 from Tier 1 Group 1 and W/E10 2.2.18 from Tier 1 Group 2. In place using manual random sampling (i.e: tokens), 059 A2.13 was selected for Tier 2 Group 1 and 034 A2.01 was selected for Tier 2 Group 2. In the process of resampling, 059 A2.13 was drawn and rejected as noted on ES-401-4. K/A 059 A2.07 was randomly chosen in its place. Also, you will note that W/E10 2.2.18, which was deleted, is on the ES-401-4 form based on a previous rejection and resampling which is now null and void.

In addition, a list of top Probabilistic Risk Assessment (PRA) related items for Seabrook Station was utilized in the examination development process. Aspects of these items have been incorporated to provide a risk based element to the examination. Also whenever possible, site-specific issues and/or Operating Experience were included. For example, one of the simulator scenarios includes a high turbine vibration casualty, which have been an ongoing issue for the unit this operating cycle. Another example is the new FPL Fleet-Wide "Conduct of Operations" (NAP-402) standard, from which written examination questions have been developed. To the extent practical, when creating the written examination, overlap with scenarios and JPM's have been minimized to make the examination more balanced and well rounded. An example of this is where the random sampling process resulted in a number of questions relating to the Loss of Secondary Heat Sink Bleed and Feed evolution. Accordingly, Loss of Secondary Heat Sink evolutions were avoided during the development of the Operating Examination.

As required by ES-301 all NRC scenarios are new or modified from existing bank scenarios. Additionally, please note that we have applied a restrictive interpretation regarding the requirements of ES-301 Section D.5.c in that we have two at power scenarios, one low power scenario (<5%), and one reactor startup scenario (supplied as a backup at the lead examiners discretion). A draft exam schedule is provided which coincides with items described on Forms ES-301-1, 2 & 5. To the extent practical the audit examination will be administered following the same schedule to provide a 'dry run' for the licensing examination week.

LICENSED OPERATOR TRAINING PROGRAM
NRC EXAM PROPOSED
DAILY SCHEDULE

	Monday 1/31	Tuesday 2/1	Wednesday 2/2	Thursday 2/3	Friday 2/4
0600	*SCENARIO #1*	*SCENARIO #2*	*SCENARIO #3*	*Simulator JPM's/Admin JPM's in Classroom*	
0700	**SRO #1/RO#1/RO#2**	**SRO #6/SRO#1/RO#1**	***SUR/RO#7/*SUR**	**JPM a & f will be run simultaneously from 0700 to 1030. This allows 15 minutes for each candidate to conduct each JPM.**	**JPM c & h will be run simultaneously from 0700 to 1030. This allows 15 minutes for each candidate to conduct each JPM.**
0800			***SUR/SRO#4/*SUR**		
0900	**SRO #2/RO#3/RO#4**	**SRO #5/RO#2/RO#3**			
1000			***SUR/SRO#6/*SUR**		
1100	**LUNCH**	**LUNCH**		**LUNCH**	**LUNCH**
1200	**SRO #3/RO#5/RO#6**	***SUR/SRO#2/RO#5**	***SUR/RO#4/*SUR**	**JPM b & g will be run simultaneously from 1100 to 1500. This allows again 15 minutes per candidate.**	**JPM d (both SRO & RO) & e (RO's only) will be run 1100 to 1500. This allows 15 minutes per candidate.**
1300			***SUR/RO#6/*SUR**		
1400	**SRO #4/SRO#5/RO#7**	***SUR/SRO#3/*SUR**			
1500			**In-Plant JPMs** ALL DAY –When not involved in Simulator	ADMIN JPM's will be conducted in a classroom concurrently while simulator JPMs are being conducted as evaluators and students are available	NRC OUT BRIEF
1600					
1700					
1800	*Note: Positions are shown in US/ATC/BOP*	*Note: Positions are shown in US/ATC/BOP *SUR is a Surrogate*	*Note: Positions are shown in US/ATC/BOP *SUR is a Surrogate*		
1900					

Written Examination administered Friday January 28th at 0700.

Facility: SEABROOK Printed: 10/29/2004

Date Of Exam: 01/31/2005

Tier	Group	RO K/A Category Points												SRO-Only Points				
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	K	A	A2	G*	
1. Emergency & Abnormal Plant Evolutions	1	3	3	3				3	3			3	18	0	0	0	0	0
	2	2	1	1				1	2			2	9	0	0	0	0	0
	Tier Totals	5	4	4				4	5			5	27	0	0	0	0	0
2. Plant Systems	1	2	2	3	3	3	3	2	2	2	3	3	28	0	0	0	0	0
	2	1	1	1	1	1	1	1	1	0	1	1	10	0	0	0	0	0
	Tier Totals	3	3	4	4	4	4	3	3	2	4	4	38	0	0	0	0	0
3. Generic Knowledge And Abilities Categories		1			2			3			4		10	1	2	3	4	0
		3			2			3			2			0	0	0	0	

Note:

1. Ensure that at least two topics from every K/A category are sampled within each tier of the RO outline (i.e., the "Tier Totals" in each K/A category shall not be less than two). Refer to Section D.1.c for additional guidance regarding the SRO sampling.
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. Select topics from many systems and evolutions; avoid selecting more than two K/A topics from a given system unless they relate to plant-specific priorities.
4. Systems/evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category /tier.
6.* 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. The SRO K/As must also be linked to 10 CFR 55.43 or an SRO-level learning objective.
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IR) 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 columns labeled "K" and "A". Use duplicate pages for RO and SRO-only exams.
8. For Tier 3, enter the K/A numbers, descriptions, importance ratings, and point totals on Form ES-401-3.
9. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

Notes shown on this page are from Draft Rev. 9. Please refer to NUREG 1021 Rev 9 Form ES-401-2 for applicable Notes.

11/10/04

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000007 Reactor Trip - Stabilization - Recovery / 1						X	2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.	3.7	1
000008 Pressurizer Vapor Space Accident / 3			X				AK3.03 - Actions contained in EOP for PZR vapor space accident/LOCA	4.1	1
000009 Small Break LOCA / 3					X		EA2.04 - PZR level	3.8	1
000011 Large Break LOCA / 3		X					EK2.02 - Pumps	2.6*	1
000015/000017 RCP Malfunctions / 4		X					AK2.07 - RCP seals	2.9	1
000022 Loss of Rx Coolant Makeup / 2			X				AK3.03 - Performance of lineup to establish excess letdown after determining need	3.1*	1
000025 Loss of RHR System / 4		X					AK2.02 - LPI or Decay Heat Removal/RHR pumps	3.2*	1
000026 Loss of Component Cooling Water / 8					X		AA2.06 - The length of time after the loss of CCW flow to a component before that component may be damaged	2.8*	1
000027 Pressurizer Pressure Control System Malfunction / 3	X						AK1.02 - Expansion of liquids as temperature increases	2.8	1
000029 ATWS / 1				X			EA1.15 - AFW System	4.1	1
000038 Steam Gen. Tube Rupture / 3					X		EA2.14 - Magnitude of atmospheric radioactive release if cooldown must be completed using steam dumps or if atmospheric reliefs lift	3.3*	1
000040 Steam Line Rupture - Excessive Heat Transfer / 4						X	2.4.4 - Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.0	1
000054 Loss of Main Feedwater / 4				X			AA1.04 - HPI, under total feedwater loss conditions	4.4	1
000058 Loss of DC Power / 6	X						AK1.01 - Battery charger equipment and instrumentation	2.8	1
000062 Loss of Nuclear Svc Water / 4						X	2.4.11 - Knowledge of abnormal condition procedures.	3.4	1
000065 Loss of Instrument Air / 8				X			AA1.03 - Restoration of systems served by instrument air when pressure is regained	2.9	1

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4			X				EK3.3 - Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations	4.0	1
W/E11 Loss of Emergency Coolant Recirc. / 4	X						EK1.1 - Components, capacity, and function of emergency systems	3.7	1
K/A Category Totals:	**3**	**3**	**3**	**3**	**3**	**3**	**Group Point Total:**		**18**

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000003 Dropped Control Rod / 1	X						AK1.03 - Relationship of reactivity and reactor power to rod movement	3.5	1
000028 Pressurizer Level Malfunction / 2				X			AA1.08 - Selection of an alternate PZR level channel if one has failed	3.7	1
000032 Loss of Source Range NI / 7						X	2.1.20 - Ability to execute procedure steps.	4.3	1
000059 Accidental Liquid RadWaste Rel. / 9					X		AA2.05 - The occurrence of automatic safety actions as a result of a high PRM system signal	3.6	1
W/E09 Natural Circ. / 4						X	2.4.12 - Knowledge of general operating crew responsibilities during emergency operations.	3.4	1
W/E10 Natural Circ. / 4					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.2	1
W/E14 Loss of CTMT Integrity / 5			X				EK3.2 - Normal, abnormal and emergency operating procedures associated with High Containment Pressure	3.1	1
W/E15 Containment Flooding / 5	X						EK1.3 - Annunciators and conditions indicating signals, and remedial actions associated with the Containment Flooding	2.8	1
W/E16 High Containment Radiation / 9		X					EK2.1 - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.0	1
K/A Category Totals:	**2**	**1**	**1**	**1**	**2**	**2**	**Group Point Total:**		**9**

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump	X											K1.10 - RCS	3.0	1
004 Chemical and Volume Control					X							K5.06 - Concept of boron "worth" or inverse boron "worth" (reactivity, pcm/ppm)	3.0	1
005 Residual Heat Removal						X						K6.03 - RHR heat exchanger	2.5	1
006 Emergency Core Cooling						X						K6.03 - Safety Injection Pumps	3.6	1
007 Pressurizer Relief/Quench Tank			X									K3.01 - Containment	3.3	1
008 Component Cooling Water		X										K2.02 - CCW pump, including emergency backup	3.0*	1
008 Component Cooling Water										X		A4.07 - Control of minimum level in the CCWS surge tank	2.9*	1
010 Pressurizer Pressure Control									X			A3.02 - PZR pressure	3.6	1
010 Pressurizer Pressure Control						X						K6.03 - PZR sprays and heaters	3.2	1
012 Reactor Protection	X											K1.01 - 120V vital/instrument power system	3.4	1
013 Engineered Safety Features Actuation					X							K5.02 - Safety system logic and reliability	2.9	1
013 Engineered Safety Features Actuation										X		A4.03 - ESFAS initiation	4.5	1
022 Containment Cooling				X								K4.03 - Automatic containment isolation	3.6*	1
022 Containment Cooling											X	2.1.33 - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	3.4	1
026 Containment Spray							X					A1.01 - Containment pressure	3.9	1
039 Main and Reheat Steam					X							K5.08 - Effect of steam removal on reactivity	3.6	1
059 Main Feedwater									X			A3.02 - Programmed levels of the S/G	2.9	1
061 Auxiliary/Emergency Feedwater				X								K4.14 - AFW automatic isolation	3.5*	1
061 Auxiliary/Emergency Feedwater											X	2.1.32 - Ability to explain and apply all system limits and precautions.	3.4	1
062 AC Electrical Distribution								X				A2.09 - Consequences of exceeding current limitations	2.7	1

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
062 AC Electrical Distribution											X	2.4.1 - Knowledge of EOP entry conditions and immediate action steps.	4.3	1
063 DC Electrical Distribution			X									K3.02 - Components using DC control power	3.5	1
064 Emergency Diesel Generator								X				A2.04 - Unloading prior to securing an ED/G	2.7	1
073 Process Radiation Monitoring							X					A1.01 - Radiation levels	3.2	1
073 Process Radiation Monitoring										X		A4.01 - Effluent release	3.9	1
076 Service Water		X										K2.01 - Service water	2.7*	1
078 Instrument Air				X								K4.02 - Cross-over to other air systems	3.2	1
103 Containment			X									K3.02 - Loss of containment integrity under normal operations	3.8	1
K/A Category Totals:	**2**	**2**	**3**	**3**	**3**	**3**	**2**	**2**	**2**	**3**	**3**	**Group Point Total:**		**28**

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Plant Systems - Tier 2 / Group 2** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
002 Reactor Coolant			X									K3.03 - Containment	4.2	1
014 Rod Position Indication								X				A2.02 - Loss of power to the RPIS	3.1	1
017 In-core Temperature Monitor											X	2.4.24 - Knowledge of loss of cooling water procedures.	3.3	1
028 Hydrogen Recombiner and Purge Control		X										K2.01 - Hydrogen recombiners	2.5*	1
033 Spent Fuel Pool Cooling							X					A1.01 - Spent fuel pool water level	2.7	1
041 Steam Dump/Turbine Bypass Control										X		A4.07 - Remote gagging of stuck open-relief valves	2.9*	1
045 Main Turbine Generator				X								K4.13 - Overspeed protection	2.6*	1
056 Condensate	X											K1.03 - MFW	2.6*	1
068 Liquid Radwaste						X						K6.10 - Radiation monitors	2.5	1
086 Fire Protection					X							K5.04 - Hazards to personnel as a result of fire type and methods of protection	2.9	1
K/A Category Totals:	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**0**	**1**	**1**	**Group Point Total:**		**10**

Generic Knowledge and Abilities Outline (Tier 3)

PWR RO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.7	1
	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	3.7	1
	2.1.28	Knowledge of the purpose and function of major system components and controls.	3.2	1
		Category Total:		**3**
Equipment Control	2.2.12	Knowledge of surveillance procedures.	3.0	1
	2.2.23	Ability to track limiting conditions for operations.	2.6	1
		Category Total:		**2**
Radiation Control	2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	2.6	1
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	2.5	1
	2.3.11	Ability to control radiation releases.	2.7	1
		Category Total:		**3**
Emergency Procedures/Plan	2.4.48	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	3.5	1
	2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	1
		Category Total:		**2**

Generic Total: **10**

Facility: SEABROOK Printed: 10/29/2004

Date Of Exam: 01/31/2005

Tier	Group	RO K/A Category Points												SRO-Only Points				
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	K	A	A2	G*	
1. Emergency & Abnormal Plant Evolutions	1	0	0	0				0	0			0	0	0	0	3	3	6
	2	0	0	0				0	0			0	0	0	0	3	1	4
	Tier Totals	0	0	0				0	0			0	0	0	0	6	4	10
2. Plant Systems	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	2	5
	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	3
	Tier Totals	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	3	8
3. Generic Knowledge And Abilities Categories		1			2			3			4		0	1	2	3	4	7
		0			0			0			0			2	2	1	2	

Note:

1. Ensure that at least two topics from every K/A category are sampled within each tier of the RO outline (i.e., the "Tier Totals" in each K/A category shall not be less than two). Refer to Section D.1.c for additional guidance regarding the SRO sampling.
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. Select topics from many systems and evolutions; avoid selecting more than two K/A topics from a given system unless they relate to plant-specific priorities.
4. Systems/evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category /tier.
6.* 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. The SRO K/As must also be linked to 10 CFR 55.43 or an SRO-level learning objective.
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IR) 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 columns labeled "K" and "A". Use duplicate pages for RO and SRO-only exams.
8. For Tier 3, enter the K/A numbers, descriptions, importance ratings, and point totals on Form ES-401-3.
9. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

Notes shown on this page are from Draft Rev. 9. Please refer to NUREG 1021 Rev. 9. form ES-401-2 for applicable notes

11/16/04

PWR SRO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000009 Small Break LOCA / 3					X		EA2.04 - PZR level	4.0	1
000015/000017 RCP Malfunctions / 4					X		AA2.11 - When to jog RCPs during ICC	3.8*	1
000022 Loss of Rx Coolant Makeup / 2						X	2.4.44 - Knowledge of emergency plan protective action recommendations.	4.0	1
000027 Pressurizer Pressure Control System Malfunction / 3					X		AA2.15 - Actions to be taken if PZR pressure instrument fails high	4.0	1
000056 Loss of Off-site Power / 6						X	2.4.7 - Knowledge of event based EOP mitigation strategies.	3.8	1
W/E11 Loss of Emergency Coolant Recirc. / 4						X	2.1.9 - Ability to direct personnel activities inside the control room.	4.0	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**3**	**3**	**Group Point Total:**		**6**

PWR SRO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000005 Inoperable/Stuck Control Rod / 1						X	2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	1
000037 Steam Generator Tube Leak / 3					X		AA2.16 - Pressure at which to maintain RCS during S/G cooldown	4.3	1
000068 Control Room Evac. / 8					X		AA2.01 - S/G level	4.3	1
W/E03 LOCA Cooldown - Depress. / 4					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.2	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**3**	**1**	**Group Point Total:**		**4**

PWR SRO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump											X	2.2.25 - Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	3.7	1
004 Chemical and Volume Control								X				A2.06 - Inadvertent boration/dilution	4.3	1
013 Engineered Safety Features Actuation								X				A2.06 - Inadvertent ESFAS actuation	4.0	1
059 Main Feedwater								X				A2.07 - Tripping of MFW pump turbine	3.3*	1
061 Auxiliary/Emergency Feedwater											X	2.1.32 - Ability to explain and apply all system limits and precautions.	3.8	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**3**	**0**	**0**	**2**	**Group Point Total:**		**5**

PWR SRO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

ES - 401 **Plant Systems - Tier 2 / Group 2** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
001 Control Rod Drive											X	2.1.22 - Ability to determine Mode of Operation.	3.3	1
015 Nuclear Instrumentation								X				A2.02 - Faulty or erratic operation of detectors or compensating components	3.5*	1
034 Fuel Handling Equipment								X				A2.01 - Dropped fuel element	4.4	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**2**	**0**	**0**	**1**	**Group Point Total:**		**3**

Generic Knowledge and Abilities Outline (Tier 3)

PWR SRO Examination Outline

Printed: 10/29/2004

Facility: SEABROOK

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.12	Ability to apply technical specifications for a system.	4.0	1
	2.1.25	Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.	3.1	1
		Category Total:		**2**
Equipment Control	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.1	1
	2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	3.7	1
		Category Total:		**2**
Radiation Control	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	3.3	1
		Category Total:		**1**
Emergency Procedures/Plan	2.4.11	Knowledge of abnormal condition procedures.	3.6	1
	2.4.15	Knowledge of communications procedures associated with EOP implementation.	3.5	1
		Category Total:		**2**

Generic Total: **7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	029 EA1.04	RO Exam: BIT inlet valve switches are not applicable to Seabrook Station, therefore resampling occurred and K/A 029 EA1.15 was selected as a replacement.
1/1	062 GEN 2.2.28	RO Exam: Loss of new and spent fuel movement procedures does not correlate well with Loss of Nuclear Service Water, therefore resampling occurred and K/A 062 GEN 2.4.11 was selected as a replacement.
1/2	W/E09 2.2.33	RO Exam: Knowledge of rod control programming does not correlate well with Natural Circulation Operations, therefore resampling occurred and K/A W/E09 2.4.12 was selected as a replacement.
2/1	022 GEN 2.1.16	RO Exam: Ability to operate plant phone, paging system, and two-way radio did not correlate well with Containment Cooling System, therefore resampling occurred and K/A 022 GEN 2.1.33 was selected as a replacement.
2/1	061 K4.09	RO Exam: Crossties between multi-unit stations is not applicable to Seabrook Station, therefore resampling occurred and K/A 061 K4.14 was selected as a replacement.
2/2	045 K4.08	RO Exam: The reactor bailey station and reactor diamond station in integrated control circuitry is not applicable to Seabrook Station, therefore resampling occurred and K/A 045 K4.13 was selected as a replacement.
1/2	W/E10 2.3.10	SRO Exam: This K/A is redundant to that of Tier 3 on the SRO exam and more relevant to that section of the exam versus Natural Circulation with Steam Void in Vessel with or without RVLIS, therefore resampling occurred and K/A W/E10 2.2.18 was selected.
2/1	059 A2.13	SRO Exam: Loss of Condensate/heater flow was manually randomly selected. Since it has a K/A value <2.5, I removed the remaining K/A's in this category < 2.5 and resampled selecting A2.07.

Seabrook Station		Date of Examination: 1/31/05
Examination Level (circle one): RO		Operating Test Number:
Administrative Topic (see Note)	**Type Code***	**Describe activity to be performed**
Conduct of Operations	M/S+	(L0027J) Perform a manual (QPTR) Calculation
Conduct of Operations	D/S+	(L0043J) Perform a Shutdown Margin Calculation (SDM) (Mode 1)
Equipment Control	P/D/S	(L0002I) Perform AC Power Source Weekly Surveillance
Radiation Control	N/S+	(LOIT04) Initiate a Liquid Effluent Sample Request.
Emergency Plan	N/A	N/A

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

*Types Codes & Criteria:

(C)ontrol Room

(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)

(N)ew or (M)odified from bank (≥ 1)

(P)revious 2 exams; (≤ 1; randomly selected)

(S)imulator +(may also be performed in classroom setting)

Seabrook Station		Date of Examination: 1/31/05
Examination Level (circle one): SRO		Operating Test Number:
Administrative Topic (see Note)	**Type Code***	**Describe activity to be performed**
Conduct of Operations	M/S+	(L0105J) Apply Technical Specifications and determine allowed outage time and mode restrictions.
Conduct of Operations	N/S+	(LOIT05) Evaluate the outcome of a Shutdown Margin Calculation (SDM) and determine impact on plant operations.
Equipment Control	N/S+	(LOIT06) Identify required notifications of on-site and off-site personnel for an off-normal event.
Radiation Control	N/S+	(LOIT07) Review and authorize release of liquid waste.
Emergency Plan	N/S	(LOIT08) Post Scenario EAL Determination and Event Classification

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.

*Types Codes & Criteria:

(C)ontrol Room

(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)

(N)ew or (M)odified from bank (≥ 1)

(P)revious 2 exams; (≤ 1; randomly selected)

(S)imulator +(may also be performed in classroom setting)

Seabrook Station	Date of Examination: **01/31/05**
Examination Level (circle one): **RO**	Operating Test Number:

Control Room Systems @ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)		
System / JPM Title	Type Code*	Safety Function
a. 001 LOIT01, Inadvertent Rod Withdrawal during Reactor S/D	N/S/L/A	1
b. 006 L0131J, Transfer to Hot Leg Injection	M/S/A	2
c. 010 L0024J, Post LOCA PORV Operation	D/S	3
d. 002 L0059J, Post Trip Primary Plant Stabilization	D/S	4
e. 076 L0058J, Transfer Service Water to the Cooling Tower	P/D/S	4
f. 028 L0084J, Start Hydrogen Recombiner	P/D/S	5
g. 029 LOIT02, Containment On-Line Purge System Lineup	N/S/A	8
h. 068 LOIT03, Respond to Liquid Radwaste High Radiation	N/S	9
In-Plant Systems @ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
i. 063 L0022J, DC Load Shedding During ECA-0.0	D/E	6
j. 012 L0008J, Local Reactor Trip During ATWS	D/E/A	7
k. 008 L0055J, Makeup to CC Head Tank During System Leak	D/E/R/A	8

@ 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	4-6 / 4-6 / 2-3
(C)ontrol Room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(L)ow power	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Seabrook Station	Date of Examination: **01/31/05**
Examination Level (circle one): **SRO(I)**	Operating Test Number:

Control Room Systems @ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)		
System / JPM Title	Type Code*	Safety Function
a. 001 LOIT01, Inadvertent Rod Withdrawal during Reactor S/D	N/S/L/A	1
b. 006 L0131J, Transfer to Hot Leg Injection	M/S/A	2
c. 010 L0024J, Post LOCA PORV Operation	D/S	3
d. 076 L0058J, Transfer Service Water to the Cooling Tower	P/D/S	4
e. 028 L0084J, Start Hydrogen Recombiner	P/D/S	5
f. 029 LOIT02, Containment On-Line Purge System Lineup	N/S/A	8
g. 068 LOIT03, Respond to Liquid Radwaste High Radiation	N/S	9
In-Plant Systems @ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
h. 063 L0022J, DC Load Shedding During ECA-0.0	D/E	6
i. 012 L0008J, Local Reactor Trip During ATWS	D/E/A	7
j. 008 L0055J, Makeup to CC Head Tank During System Leak	D/E/R/A	8

@ 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	4-6 / 4-6 / 2-3
(C)ontrol Room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(L)ow power	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Seabrook Scenario No.: 1 Op Test No.: 1

Examiners: _____ Candidates: _____

_____ _____

_____ _____

Initial Conditions: Mode 1. Unit at 75% power. IC # 210.

Turnover: The crew will take the shift and commence a rapid downpower to 50% to remove "A" Main Feedwater Pump from service within the next hour.

Entered TSASs for CS-P-2B being tagged out of service for planned maintenance 2 hours ago. Return to service expected within 6 hours.

Critical Tasks:

1. MANUALLY trip the reactor from the control room when SSPS fails to automatically trip the reactor. [E-0]
2. Control the EFW flow rate to not less than 25 GPM per SG in order to minimize the RCS cooldown rate before a severe (orange path) challenge develops to the integrity CSF. [ECA-2.1]

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R (RO) N (BOP/US)	Rapid power decrease.
2	ptFWPT505	I (BOP/US) TS (US)	PT-505 Turbine First Stage Pressure Transmitter Fails LOW
3	ltRCLT459	I (RO/US) TS (US)	Controlling PZR Level Channel LT-459 fails LOW
4	mfTH002 mfTH002 (severity increases) mfRPS001 mfRPS002	C (BOP/US) M (ALL) C (RO/US)	Turbine Generator Vibrations begin to increase. After entry into abnormal operating procedure, turbine vibrations will rapidly increase beyond automatic turbine trip setpoint resulting in a turbine trip. The reactor fails to trip automatically when the turbine trips. The crew will have to trip the reactor manually (CT).
5	mfMS051 mfRPS019 mfRPS020 svMSV86 svMSV88 svMSV90 svMSV92 mfSI003	M (ALL) C (BOP) C (RO)	The combination of high turbine vibrations and turbine trip causes a catastrophic rupture of the main steam bottle (down stream of MSIVs). All four MSIVs will fail to close when the MSI signal is actuated. Manual actuation of MSI in the control room should be attempted, but will not cause the MSIVs to close. Upon automatic actuation of Safety Injection from the RCS cooldown, the "A" Safety Injection pump will not automatically start. Procedure progression will be E-0 ⇒ E-2 ⇒ ECA-2.1 where the crew will be directed to reduce feed flow to all SGs to 25 gpm to avoid severe challenge to the Integrity CSF (CT).
6	mvFWFV4214A	C (BOP/US)	The motor operator overloads for EFW flow control valve FW-FV-4214A will trip as soon as valve motion is demanded. The operator will be required to utilize FW-FV-4214B to control EFW flow to A SG.

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

SCENARIO 1 OVERVIEW

The crew will take the shift at 75% with instructions from the SM to reduce power to 50% within the next hour as requested by the dispatcher.

After the crew has the shift (prior to placing rods in manual for the downpower), PT-505 fails low. The crew responds in accordance with OS1235.05, "Turbine Impulse Pressure PT-505 or PT-506 Instrument Failure". The failure of this instrument will result in inward rod movement if rod control is in AUTO. The RO should place rod control in manual to halt rod insertion. The crew will BYPASS PT-505 and continue with the downpower with SM permission. The US will address TS in accordance with OS1235.05.

The crew continues with the power reduction. After the crew reduces power by 3-4%, or at the lead examiners instruction, the controlling PZR level channel, LT-459, fails low. The crew responds in accordance with OS1201.07' "PZR Level Instrument Failure". Charging flow will increase and letdown will isolate. The RO is expected to take manual control of the PZR level controller and restore PZR level. The RO will also restore letdown to service. The US will address TS in accordance with OS1202.07.

After letdown is restored and PZR level is stabilized, the main turbine will experience vibration problems. The crew responds in accordance with ON1231.01, "Turbine Generator High Vibration". After progressing beyond step 1 and at the lead examiners instruction, the turbine vibrations will increase very rapidly beyond the automatic turbine trip setpoint resulting in a Turbine Trip. Since the reactor does not automatically trip, the crew must trip the reactor (CT) and enter E-0, "Reactor Trip or Safety Injection".

The combination of high turbine vibrations and turbine trip causes a catastrophic rupture of the main steam bottle (down stream of MSIVs). All four MSIVs will fail to close when the MSI signal is actuated. Manual actuation of MSI in the control room will not cause the MSIVs to close. The "A" Safety Injection pump will not automatically start, and should be manually started as part of E-0, Attachment A actions.

Procedure progression will be E-0 ⇒ E-2, "Faulted Steam Generator Isolation" ⇒ ECA-2.1, "Uncontrolled Depressurization of All Steam Generators" where the crew will be directed to reduce feed flow to all SGs to 25 gpm (CT) to avoid severe challenge to the Integrity CSF. A component failure will occur as the operator attempts to limit EFW flow to the "A" S/G. The motor overloads for FW-FV-4214A (Train "A" EFW throttle valve to "A" S/G) will actuate and require the operator to utilize FW-FV-4214B (Train "B" EFW throttle valve to "A" S/G) to limit EFW to SG A to 25 gpm. When feed flow has been reduced, NSO(s) dispatched to the MSIVs will successfully close MSIVs in the west pipe chase (SG A and D). The crew will exit ECA-2.1 returning to E-2. With EFW limited to 25 gpm a RED condition will exist for the HEAT SINK CSF. A note at the beginning of FR-H.1, "Response to Loss of Heat Sink" states that the procedure should NOT be implemented if feed flow was reduced by operator action. The scenario will be terminated at Lead Examiner discretion.

Facility:	Seabrook	Scenario No.:	2	Op Test No.:	2
Examiners:	_____	Candidates:	_____		
	_____		_____		
	_____		_____		

Initial Conditions: Mode 2. Unit is operating at 10^{-8} amps , MOL. IC# 211

Turnover: The crew will take the shift and commence a power increase up to but not to exceed 3% power.

The "C" Primary Component Water pump is tagged out for bearing replacement. The motor experienced high vibrations during quarterly surveillance.

Seismic Event occurred two shifts ago.

Critical Tasks:

1. MANUALLY actuate at least one train of Containment Spray or start one train of Containment Spray Building Equipment prior to step 7 of E-0 following Large Break LOCA. [E-0]
2. PERFORM TRANSFER to cold leg recirculation when automatic swapover occurs or 115,000 gallons in the RWST, whichever occurs first. [ES-1.3]

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R (RO) N (BOP/US) TS (US)	Raise reactor power from 10^{-8} amps to approximately 3% power. Place Electric EFW Pump (P-37B) in PTL after SM informs crew that it is reported to be inoperable and address TS.
2	ltFWLT529	I (BOP/US) TS (US)	FW-LT-529 Fails LOW causing the "B" Feedwater Regulating Bypass valve to fail open.
3	ptRCPT455	I (RO/US) TS (US)	Controlling PZR pressure channel fails HIGH.
4	mfRC024A mfCBS004 mfCBS005 mfFCS002 mfFW039 svMS1V395 bkFWP37B	M (ALL) C (RO/US) C (RO) C (BOP)	Large Break LOCA occurs leading to automatic reactor trip with failure of Containment Spray pumps to automatically start upon Phase "B" actuation. The crew will have to manually start at least one Containment Spray pump (CT). Upon completion of immediate actions, all RCPs should be tripped based on E-0 Operator Action Summary. The "B" CCP pump will not automatically start, and should be manually started as part of E-0, Attachment A actions. The SUFP will trip after reactor trip and MS-V395 will close after the Steam Driven EFW Pump Auto Starts, causing it to shutdown. It should be recognized by the BOP that there is no EFW flow. Procedure progression will be E-0 to E-1 to FR-P.1 (Red) to FR-Z.1 (Orange) to E-1 to ES-1.3.
5	N/A	N/A	Perform transfer to cold leg recirculation per ES-1.3 when the automatic swapover occurs or 115,000 gallons in the RWST, whichever occurs first. (CT)

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

SCENARIO 2 OVERVIEW

The crew will take the watch at 10^{-8} amps on the intermediate range channels. The "C" PCCW Pump is tagged out for maintenance. The crew is instructed to continue a power increase in accordance with OS1000.07, "Approach to Criticality" and OS1000.02, "Plant Startup from Hot Standby to Minimum Load".

After the crew assumes the watch, they will continue power ascension in accordance with OS1000.07, "Approach to Criticality". Once the lead examiner is satisfied with the reactivity addition, the SM will inform the US that Engineering has determined the Electric EFW Pump to be inoperable. The US should direct P-37A placed in PTL and address TS. He should recognize that a Mode change is not allowed based on these plant conditions.

Once the US has evaluated TS for EFW pump condition and at lead examiners discretion, the "B" S/G level transmitter, LT-529, fails LOW. The crew should respond to the failure in accordance with OS1253.03. The high input to the feed control system will cause "B" S/G feed regulating bypass valves to modulate OPEN, increasing feed flow to "B" S/G. There is no redundant channel for this level transmitter thus the BOP operator must maintain manual control of S/G "B" feed regulating bypass valve for the remainder of the startup. The US will verify TS compliance at the discretion of the lead examiner during or following the scenario.

After the US verifies TS compliance or at the lead examiners discretion, the controlling PZR pressure instrument fails high. PZR spray initiates and RCS pressure will decrease rapidly. It is expected that the RO will identify the failed channel and take manual control of the master pressure controller, stabilizing RCS pressure. TS assessment is can be performed during the scenario or immediately following at the Lead Examiners discretion. If the crew is not timely, the plant may trip on low pressure safety injection (low pressure trip blocked by P-7/P-10) which will not adversely affect the scenario. Whether the crew manually trips the reactor or the reactor automatically trips, A Large Break LOCA will occur with a failure of both containment spray pumps to automatically start. It is a critical task that the crew manually starts at least one containment spray pump prior to step 7 of E-0 (CT).

Following E-0, "Reactor Trip or Safety Injection" immediate actions, the RCPs should be tripped based on E-0 Operator Action Summary required actions. It should be recognized that the "B" CCP did not start and the pump should be manually started during E-0 Attachment A actions. The BOP should recognize the SUFP tripped and the Steam Driven EFW PUMP started but MS-V395 closed and P-37A secured start post reactor trip, therefore no EFW flow exists. (Electric EFW Pump P-37B previously placed in PTL). Note that at this low power, there is still adequate S/G inventory and S/G heat sink is not an issue during large break LOCA.

Procedure progression following reactor trip is E-0 to E-1, "Loss of Reactor or Secondary Coolant" to FR-P.1, "Response to Imminent Pressurized Thermal Shock Conditions" (RED) to FR-Z.1, "Response to High Containment Pressure" (ORANGE) to E-1 and to ES-1.3, "Transfer to Cold Leg Recirculation". Note that transitions to FR-P.1 and FR-Z.1 are dependant upon whether plant conditions are met which differ during a low power scenario based on crew progression through procedures. Note that within three minutes of receiving the RWST Lo-Lo level alarm the first three steps of ES-1.3 must be performed. It is a critical task that at least one train of cold leg recirculation be established (CT). Note also that because this is a low power scenario, it is possible depending on how expeditious the crew is to start containment spray, that FRP entry conditions may NOT be met (ie: RCS temperature does may not drop low enough or containment pressure does not remain high enough).

Facility:	Seabrook	Scenario No.:	**3**	Op Test No.:	3
Examiners:	_____	Candidates:	_____		
	_____		_____		
	_____		_____		

Initial Conditions: Mode 1. Unit is operating at 100% power. IC# 212

Turnover: The ASDV for the "D" Steam Generator (MS-PV-3004) is tagged out of service due to a positioner air leak. Entered TSAS 3.3.3.5 action c, 3.6.3 action c and 3.7.1.6 action a, two hours ago. Expected outage time is 12 hours.

The Pressurizer control group heaters are tagged out of service. The control circuit for the heaters has failed to zero output. A troubleshooting plan has been developed and Electrical Maintenance is investigating the problem. The heaters have been out of service for 10 hours. Backup Heater Group "B" is ON.

Critical Tasks:

1. MANUALLY trip the reactor from the control room when SSPS fails to automatically trip the reactor. [E-0]
2. MANUALLY trip the main turbine before a severe (Orange Path) challenge develops to either the Subcriticality or the Integrity CSF, or before transition to ECA-2.1, whichever happens first. [E-0]
3. MANUALLY trip the "C" and "D" RCPs when subcooling is <40 F such that an Orange path on Core Cooling does not occur when forced circulation in the RCS stops.

Event No.	Malf. No.	Event Type*	Event Description
1	mfHD027	C (BOP/US)	"A" Heater Drain Pump Trips on overcurrent
2	N/A	R (ALL)	As a result of the heater drain pump trip, a loss of feedwater preheating will result in positive reactivity and subsequent power increase. The crew will need to take positive control to restore power less than 100%.
3	ttRCTT411	I (RO/US) TS (US)	Loop 1 Tc Instrument Fails HIGH
4	mfED001 mfRPS001 mfRPS002 mfRPS003	M (ALL) C (RO/BOP)	Loss of 13.8kV Bus 1 causes a Loss of "A" & "B" RCP's with a failure of an automatic reactor trip (CT) and main turbine trip (CT) to occur. Note that although Safety Injection should actuate, ALL Safety Injection pumps will be prevented from manual start (malfunction numbers are not shown since no credit is taken for these failures)
5	mfRC016 mfRC019 mfRC049D	M (ALL) C (RO/US)	Subsequent to reactor trip, the Reactor Vessel will develop a flange leak. The crew will need to trip the remaining "C" and "D" RCPs based on loss of RCS subcooling (CT). Once tripped, the RCS leak will become significantly larger to force the crew into FR-C series procedures. The "A" CCP will also trip on overcurrent.

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

SCENARIO 3 OVERVIEW

The crew will take the watch with reactor power at 100%. The ASDV for the "D" Steam Generator (MS-PV-3004) is tagged out of service due to a positioner air leak. TSAS 3.3.3.5 action c, 3.6.3 action c and 3.7.1.6 action a was entered two hours ago. Expected outage time is 12 hours.
The Pressurizer control group heaters are tagged out of service. The control circuit for the heaters has failed to zero output. A troubleshooting plan has been developed and Electrical Maintenance is investigating the problem. The heaters have been out of service for 10 hours. Backup Heater Group "B" is ON.

After the crew assumes the watch, the "A" Heater Drain Pump will trip on overcurrent. The crew should respond to the pump trip in accordance with OS1290.02, "Response to Condensate or Feedwater Heater System Transient". As a result of this malfunction, the colder water being fed into the steam generators due to loss of preheating will result in a positive reactivity addition and resultant power increase greater than rated thermal. The crew must respond by lowering turbine load.

Once power is restored below rated thermal and the crew has performed the first few steps of OS1290.02 or at the lead examiner's discretion, the Loop #1 narrow range temperature instrument will fail HIGH. This results in Tavg for Loop #1 failing high and an automatic rod insertion. The RO should take manual control of control rods and stop uncontrolled insertion. The crew will respond to the Tc instrument failure in accordance with OS1201.08, "Tavg/Delta T Instrument Failure". The crew should defeat affected loop delta-T and Tavg inputs. The US should verify TS compliance.

At the discretion of the lead examiner, A Loss of 13.8kV Bus 1 will occur. The 13.8 kV Bus 1 UAT breaker trips and the associated RAT breaker fails to fast transfer. Loss of power to Bus 1 results in reactor trip demand due to loss off A & B RCPs and A & C Circulating Water pumps. The reactor fails to trip automatically and the operator is required to manually trip the reactor (CT). On the reactor trip, the automatic Turbine Trip fails. The operator is required to manually trip the main turbine (CT). The crew will enter E-0, "Reactor Trip or Safety Injection".

Upon reactor trip, the Reactor Vessel Flange will develop a 275 gpm leak resulting in a SBLOCA. (Note to speed up the scenario and drop subcooling quicker, leak size will be increased greater than 275 gpm) This will result in Safety Injection actuation which will automatically or manually be actuated, however no SI pumps will start nor will manual starts be successful from the control room. Note that safety injection may also occur due to the cooldown associated with the Main Turbine not tripping depending on how expeditious the operators are in tripping the Main Turbine. Also note that all attempts to establish Safety Injection will NOT be successful as the intention is to force the crew into FR-C.1, "Response to Inadequate Core Cooling". The "A" CCP will trip on overcurrent following reactor trip. Because of the loss of subcooling, the crew is required by the Operator Action Summary of E-0 to trip the remaining running RCP's (C & D) (CT). Once RCP's are tripped the SBLOCA will become significantly larger to drive the crew into inadequate core cooling FRP's.

Procedure progression following reactor trip is E-0 to ES-0.1, "Reactor Trip Response" and back to E-0 to E-1, "Loss of Reactor or Secondary Coolant" potentially to ECA-1.1, "Loss of Emergency Coolant Recirculation" and then to FR-C.2, "Response to Degraded Core Cooling" (ORANGE) and/or FR-C.1 (RED) when conditions are met. The scenario will end when the crew recognizes the positive displacement charging pump cannot be started because the mini-flow recirculation valve (CS-V-205) cannot be opened and/or maintenance gives the crew back one of the SI pumps or at the discretion of the lead examiner.

Facility:	Seabrook	Scenario No.:	4	Op Test No.:	Spare if needed
Examiners:	_____	Candidates:	_____		
	_____		_____		
	_____		_____		

Initial Conditions: Mode 3. Reactor Startup in progress. Shutdown Banks have been withdrawn. IC# 213

Turnover: The crew will take the shift and continue with reactor startup at step 4.4 of OS1000.07. PZR Sprays are being forced using two banks of backup heaters. Startup Feedwater Pump is running powered from Bus 4 supplying S/G's via Main Feedwater Regulating Bypass Valves. All MSIV's are open and condenser vacuum has been established. Steam Dump control is in the Steam Pressure Mode.

TSAS 3.1.1.4 (monitoring RCS temperature) has been entered and being monitored every 15 minutes.

No equipment is out of service and Mode 2 checklist and TS logs have been completed.

Severe Weather (high winds) is anticipated sometime during your shift.

Critical Tasks:

1. MANUALLY actuate at least one train of SI actuated safeguards before transition to any E-1, E-2, E-3 or FRP. [E-0]
2. ISOLATE feedwater flow into and steam flow from the ruptured S/G before a transition to ECA-3.1 is required. [E-3]

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R (ALL) N (ALL) TS (US)	Withdraw control bank rods during reactor startup.
2	mfCS016	C (RO/US) TS (US)	Trip of Centrifugal Charging Pump CS-P2A
3	mfAR005A	C (BOP/US)	Trip of "A" Mechanical Vacuum Pump
4	mfED038 cSWV16 mfED033 mfSG002A	M (ALL) C (BOP/US) M (ALL)	Offsite power is lost resulting in a reactor trip. The "A" EDG will start and load onto Emergency Bus 5, however, its associated SW cooling valve (SW-V16) will not automatically open and must be manually opened. The "B" EDG will fail to start. Upon reactor trip, a large Steam Generator Tube Rupture in the "A" S/G at 500 gpm will occur.
5	mfMS049A mfRPS007 mfRPS008	M (ALL) I (RO/US)	RCS pressure will continue to drop post trip due to SGTR and the crew must manually actuate Safety Injection (CT) due to failure of automatic SI. After SI actuation, the "A" S/G will fault inside containment. Procedure progression will be E-0 to E-2 to E-3 to ECA-3.1.

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

SCENARIO 4 OVERVIEW

Note that this is a backup scenario or can be used in lieu of any of the other scenarios at Lead Examiners discretion.

The crew will take the shift with a reactor startup in progress currently at step 4.4 of OS1000.07. All shutdown banks of control rods have been withdrawn.

After the crew assumes the watch, they will continue the reactor startup be withdrawing control banks and plotting 1/M data in accordance with OS1000.07, "Approach to Criticality". Once the lead examiner is satisfied with the reactivity addition and TS compliance with the reactor startup, the "A" CCP will trip on overcurrent. The crew should respond to the failure in accordance with OS1202.02, "Charging System Failure". The crew should also establish priorities regarding reactor startup and recognize that a mode change is not allowed. Letdown will be isolated by closing CS-V-145 and the "B" CCP will be started. The US will verify TS compliance at the discretion of the lead examiner during or following the scenario.

Once letdown is restored or at the discretion of the lead examiner, the running vacuum priming pump (P-50A) will trip. The should reference the VAS alarm procedure (F5117) and start a standby pump. The crew may also respond in accordance with ON1233.01, "Loss of Condenser Vacuum". This procedure has the crew manually start the standby mechanical vacuum pumps, and provides criteria to trip the plant which should not be necessary as starting the standby pump will restore vacuum. Note condenser vacuum is not adversely affected at this power level by a loss of one vacuum priming pump.

At the lead examiners discretion, a Loss of Off-Site Power will occur. A subsequent reactor trip will automatically occur. The "A" EDG will automatically start and load onto Bus 5, however it's associated SW cooling valve (SW-V16) will not automatically open. The crew must recognize that this valve is closed and open it to prevent overheating the only running EDG. The "B" EDG will not automatically or manually start, resulting in a loss of power to Bus 6. A loss of charging and subsequent letdown (note letdown may still be isolated based on examiner discretion) will occur due to previous failure of the "A" CCP.

Upon reactor trip, a Steam Generator Tube Rupture will occur at 500 gpm. The crew must manually actuate Safety Injection (CT), because automatic SI has failed. Upon SI actuation, a fault in the "A" Steam Generator will occur inside containment

Procedure progression following reactor trip is E-0, "Reactor Trip or Safety Injection" to E-2, "Faulted Steam Generator Isolation" to E-3, "Steam Generator Tube Rupture" to ECA-3.1, "SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired". (May transition to ES-0.1, "Reactor Trip Response" for a short time until SGTL progresses and SI criteria are met) Steam flow must be secured to the steam driven feedwater pump from the ruptured Steam Generator (A) and feedwater flow must also be secured (CT) prior to transition to ECA-3.1.