Commonwealth Edison Company
Byron Generating Station
4450 North German Church Road
Byron, IL 61010-9794
Tel 815-234-5441

June 23, 1998

ComEd

LTR: Byron 98-0195
FILE: 1.06.5110

Mr. Hironori Peterson
U.S. Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

Dear Mr. Peterson:

Enclosed is the integrated examination outline (written and operating test) which Byron Generating Station is submitting for review, comment, and approval for the Initial License Examination scheduled for the week of September 14, 1998, at Byron Generating Station.

This outline has been developed in accordance with Interim Revision 8 of NUREG-1021 ("Operator Licensing Examiner Standards").

Please ensure that these materials are withheld from public disclosure, until after the examinations are complete.

If you have any questions or concerns regarding this outline, please contact Mick Brown at (815) 234-5441 extension 3133.

Sincerely,

K.C. Kofen
for Ken Graesser
Site Vice-President
Byron Nuclear Generating Station

cc: No Enclosures
Regulatory Assurance
T. Gierich
T. Schmidt
E. Bendis
S. Pettinger
P. Hippely
R. Franklin
R. Brown
Class File

Examination Outline
Quality Assurance Checklist

Facility: BYRON UNIT 1 AND 2 Date of Examination: SEPT 14, 1998

Item	Task Description	Initials a	Initials b	Initials c	
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model per ES-401.	RGB	DFF	HP	
	b. Assess whether all six knowledge and four ability categories are appropriately sampled.	RGB	DFF	HP	
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	RGB	DFF	HP	
	d. Assess whether the repetition from previous examination outlines is excessive.	RGB	DFF	HP	
2. SIM	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, and major transients.	RGB	DFF	HP	* Note B
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity; ensure each applicant can be tested using at least one new scenario and scenarios will not be repeated over successive days.	RGB	DFF	HP	
	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.	RGB	DFF	HP	* Note ~~B~~C
3. W/T	a. Verify that the outline(s) contain(s) the required number of control room and in-plant tasks and verify that no more than 30% of the test material is repeated from the last NRC examination.	RGB	DFF	HP	
	b. Verify that the tasks are distributed among the safety function groupings as specified in ES-301; one task shall require a low-power or shutdown condition, one or two shall require the applicant to implement an alternate path procedure, and one should require entry to the RCA.	RGB	DFF	HP	
	c. Verify that the required administrative topics are covered, with emphasis on performance-based activities.	RGB	DFF	HP	
	d. Determine if there are enough different outlines to test the projected number and mix of applicants and ensure that no more than 30% of the items are duplicated on successive days.	RGB	DFF	HP	
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam section.	RGB	DFF	HP	
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.	RGB	DFF	HP	
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	RGB	DFF	HP	
	d. Check for duplication and overlap among exam sections.	RGB	DFF	HP	
	e. Check the entire exam for balance of coverage.	RGB	DFF	HP	* Note D
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	RGB	DFF	HP	

	Printed Name / Signature	Date
a. Author	RONALD G. BROWN / RG Brown	6/24/98
b. Facility Reviewer(*)	David F. Flowers / D F Flowers	6/24/98
c. Chief Examiner	HIRONORI Peterson / H Peterson	7/16/98
d. NRC Supervisor	MELVYN LEACH / Melvyn Leach	8/7/98

} NOTE A.

(*) Not applicable for NRC-developed examinations

NOTE A: QA sheet signed after cover letter date!?

NOTE B: See ES-301-5 - License [illegible] cannot take one event (power increase) as both (N) & (R). Also malfunction counted as only one (I) or (C) - not both

NOTE C: NO-301-4 ~~for~~ the outline submittal. Licensee reminded to ensure submittal of QA Form ES-301-4 with material submittal

NOTE D: Suggested to increase areas with only one item to at least two to ensure balance. on ES-401-3

Facility: Byron 1 & 2 Examination Level: RO		Date of Examination: September 14, 1998 Operating Test Number: 1
Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A1	Plant Parameter Verification / Complete An Estimated Critical Condition Checklist	1. JPM K/A 2.1.7 3.7/4.4
	Security Familiarity / Respond To Telephoned Bomb Threat	1. JPM K/A 2.4.28 2.3/3.3
A2	Clearance And Tagging / Identify And Replace Blown Fuse	1. JPM K/A 2.2.13 3.6/3.8
A.3	Protection From Radiation Exposure / Prepare For Entry Into High Radiation Area > 1000 mr/hr	1. JPM K/A 2.3.10 2.9/3.3
A.4	Emergency Plan / Emergency Plan Directions	2. a. K/A 2.3.1 2.6/3.0 Emergency Exposures
		2. b. K/A 2.4.29 2.6/4.0 Emergency facilities

Facility: Byron 1 & 2 Examination Level: SRO-I		Date of Examination: September14, 1998 Operating Test Number: 1
Administrative Topic/Subject Description		Describe method of evaluation: 3. ONE Administrative JPM, OR 4. TWO Administrative Questions
A1	Plant Parameter Verification / Review A Completed Estimated Critical Condition	1. JPM K/A 2.1.7 3.7/4.4
	Conduct of Operations / Shift Turnover with Staffing Complications	1. JPM K/A 2.1.3 3.0/3.4; 2.1.4 2.3/3.4
A2	Clearance And Tagging / Identify And Replace Blown Fuse	1. JPM K/A 2.2.13 3.6/3.8
A.3	Protection From Radiation Exposure / Prepare For Entry Into High Radiation Area > 1000 mr/hr	1. JPM K/A 2.3.10 2.9/3.3
A.4	Emergency Plan / GSEP Classification And Protective Action Recommenadtions	1. JPM K/A 2.4.38 2.2/4.0

Facility: Byron 1 & 2 Examination Level: SRO-I		Date of Examination: September 14, 1998 Operating Test Number: 1
System / JPM Title / Type Codes*	Safety Function	Planned Followup Questions: K/A/G – Importance - Description
1. Nuclear Instrumentation / Response to Source Range NI failure - MODE 5 N S L	VII	a. 015000K505 4.1/4.5 Determination of criticality b. 015000A102 3.5/3.6 Evaluation for flux doubling
2. Rod Control System / Rod Cluster Control Exercise(N-41) D S	I	a. 001000K402 3.8/3.8 Startup Reset pushbutton response b. 001000A302 3.7/3.6 Rod Insertion Limits
3. RCS Pressure Control / Depressurize the RCS per ES-1.2 Post-LOCA Cooldown and Depressurization N A S	III	a. 002000K105 3.2/3.4 PRT Pressure greater than RCDT pressure b. 000008A108 3.8/3.8 Effect of leaking safety of increase in PRT pressure
4. Containment Spray / Manual CS Actuation N A S	V	a. 026000A105 3.1/3.4 Spray Additive Tank eductor flow b. 026000A401 4.5/4.3 AUTO start conditions for CS Pump
5. CVCS / Establish Excess Letdown to the RCDT(N-11) M S	II	a. 004000A212 4.1/4.3 Effect of SI on excess letdown/seal return b. 004000K104 3.4/3.8 Impact of excess letdown (high pressure) on RCPs
6. AC Electrical / Establish Shutdown Electrical Lineup N S	VI	a. 062000K403 2.8/3.1 ACB interlock configuration (Auto transfer setup) b. 062000A206 3.4*/3.9 Aligning bus with SAT to be deenergized (cross-ties)
7. Reactor Coolant System / Start a Reactor Coolant Pump (0030390101.1 M) N S L	IV	a. 003000A303 3.2/3.1 #2 seal indications on RCP startup b. 012000A306 3.7/3.7 Reactor trip on loss of RCP
8. Secondary Heat Removal - AFW / Local Emergency Start of 1B AFW Pump (N-56mod) M A P R	IV	a. 061000A104 3.9/3.9 AF Suction pressure protection b. 061000K302 4.2/4.4 AF required flow
9. Diesel Generator System / Local Abnormal Start of a D/G (N-35b) D P	VI	a. 064000K105 3.4/3.9 Starting Air Compressor operation b. 064000A406 3.9/3.9 Emergency stopping of EDG
10. Component Cooling Water System / Locally align U-1 CC System for Post-LOCA Condition - Train Separation N P R	VIII	a. 008000A203 3.0/3.2 Maximum CCW Heat Exchanger outlet Temperature b. 008000A303 3.0/3.1 CCW system minimum flow requirements
Type Codes: (D) Direct from bank, (M)odified from bank, (N)ew, (A)lternate Path, (C)ontrol Room, (S)imulator, (P)lant, (L)ow Power, (R)CA		

Facility: Byron 1 & 2 Examination Level: RO	Date of Examination: September 14, 1998 Operating Test Number: 1

System / JPM Title / Type Codes*	Safety Function	Planned Followup Questions: K/A/G – Importance - Description
1. Nuclear Instrumentation / Response to Source Range NI failure - MODE 5 N S L	VII	a. 015000K505 4.1/4.5 Determination of criticality b. 015000A102 3.5/3.6 Evaluation for flux doubling
2. Rod Control System / Rod Cluster Control Exercise(N-41) D S	I	a. 001000K402 3.8/3.8 Startup Reset pushbutton response b. 001000A302 3.7/3.6 Rod Insertion Limits
3. RCS Pressure Control / Depressurize the RCS per ES-1.2 Post-LOCA Cooldown and Depressurization N A S	III	a. 002000K105 3.2/3.4 PRT Pressure greater than RCDT pressure b. 000008A108 3.8/3.8 Effect of leaking safety of increase in PRT pressure
4. Containment Spray / Manual CS Actuation N A S	V	a. 026000A105 3.1/3.4 Spray Additive Tank eductor flow b. 026000A401 4.5/4.3 AUTO start conditions for CS Pump
5. CVCS / Establish Excess Letdown to the RCDT(N-11) M S	II	a. 004000A212 4.1/4.3 Effect of SI on excess letdown/seal return b. 004000K104 3.4/3.8 Impact of excess letdown (high pressure) on RCPs
6. AC Electrical / Establish Shutdown Electrical Lineup N S	VI	a. 062000K403 2.8/3.1 ACB interlock configuration (Auto transfer setup) b. 062000A206 3.4*/3.9 Aligning bus with SAT to be deenergized (cross-ties)
7. Reactor Coolant System / Start a Reactor Coolant Pump (0030390101.1 M) N S L	IV	a. 003000A303 3.2/3.1 #2 seal indications on RCP startup b. 012000A306 3.7/3.7 Reactor trip on loss of RCP
8. Secondary Heat Removal - AFW / Local Emergency Start of 1B AFW Pump (N-56mod) M A P R	IV	a. 061000A104 3.9/3.9 AF Suction pressure protection b. 061000K302 4.2/4.4 AF required flow
9. Diesel Generator System / Local Abnormal Start of a D/G (N-35b) D P	VI	a. 064000K105 3.4/3.9 Starting Air Compressor operation b. 064000A406 3.9/3.9 Emergency stopping of EDG
10. Component Cooling Water System / Locally align U-1 CC System for Post-LOCA Condition - Train Separation N P R	VIII	a. 008000A203 3.0/3.2 Maximum CCW Heat Exchanger outlet Temperature b. 008000A303 3.0/3.1 CCW system minimum flow requirements

Type Codes: (D) Direct from bank, (M)odified from bank, (N)ew, (A)lternate Path, (C)ontrol Room, (S)imulator, (P)lant, (L)ow Power, (R)CA

The JPMs are planned to be performed under the following conditions:

JPMs 1 & 7 are to be run in RCS low temperature and pressure conditions (MODE 5).
The desired conditions are approximately 188°F and 360 psig. This can be done with either RCS solid or Pzr bubble drawn with RCS loops filled. Also, the Shutdown Banks are fully withdrawn and Source Range channel N-31 selected as audio count rate channel.

JPM 1, SRNI N-31 failed low, requires no particular plant conditions and is therefore suited for coupling with JPM 7. The actions are performed as directed by BOA INST-1. The evaluated steps are 1, Attachment C steps 1-3, 5.

JPM 7 is normal startup of the first RCP as directed by BGP 100-1 step F.28 and directed by BOP RC-1. The steps of RC-1 should be checked through step 18 and steps marked as complete through (& to include) step 3. The evaluated actions are steps 4 through 26. **NOTE that RC-1 indicates Temporary Procedure 98-0-68 exists.**

JPMs 2, 5 & 6 are to be run in at-power condition.
The desired conditions are normal for the current power level.

JPM 2. The actions of 1BOS 1.3.1.2-1 steps F.1 - F.2 and F.5-F.6 are evaluated for Shutdown Bank E and Control Bank A.

JPM 5. The actions of 1BOP CV-15 Step F.1 are performed with excess letdown aligned to the RCDT.

JPM 6. The actions of BGP 100-4 Step F.15 are performed to align normal loads from the UATs to the SATs.

JPMs 3 & 4 are to be run in Post-LOCA condition (about 700 gpm leak).
The desired conditions are:

1. Break flow equal to injection flow with TWO CV pumps running in HHSI alignment.
2. MSIVs open (either block Steamline SI, or reset if actuated, and open MSIVs)
3. RCS pressure between 1000 psig and 1450 psig
4. CNMT pressure is expected to be ADVERSE and should indicate that at some point 20 psig was exceeded
5. Pzr level about 40% but less than 50% (level that is termination point for actions of JPM 3)
6. All RCPs stopped
7. SI reset, Phase A reset and air restored to CNMT, Phase B reset (if actuated)

JPM 3, the Pressurizer spray valves (and controllers, if required) are to be failed closed. The major steps of BEP-1 are complete through STEP 12. The steps of BEP ES-1.2 are complete through STEP 7. A cooldown should be set up as appropriate (20°F/hr to 50°F/hr). 1BEP ES-1.2 steps 9 & 10 are the evaluated actions for the JPM.

JPM 4, CNMT Spray actuation failed, CS007 valves closed (NOT failed), CS019 valves closed (NOT failed), One CS pump running, if possible. The evaluated actions are those of ATTACHMENT B of BEP-0. Would like to have one CS pump running with associated CS007 valve closed. The other pump can be NOT running. This forces the RNO actions of step 1.b. CS019 valves should be close (if this does not affect CS pump in step 1.b). One CS pump is restarted per step 1.b, if possible, and the other (or both) is started in step 2.a.

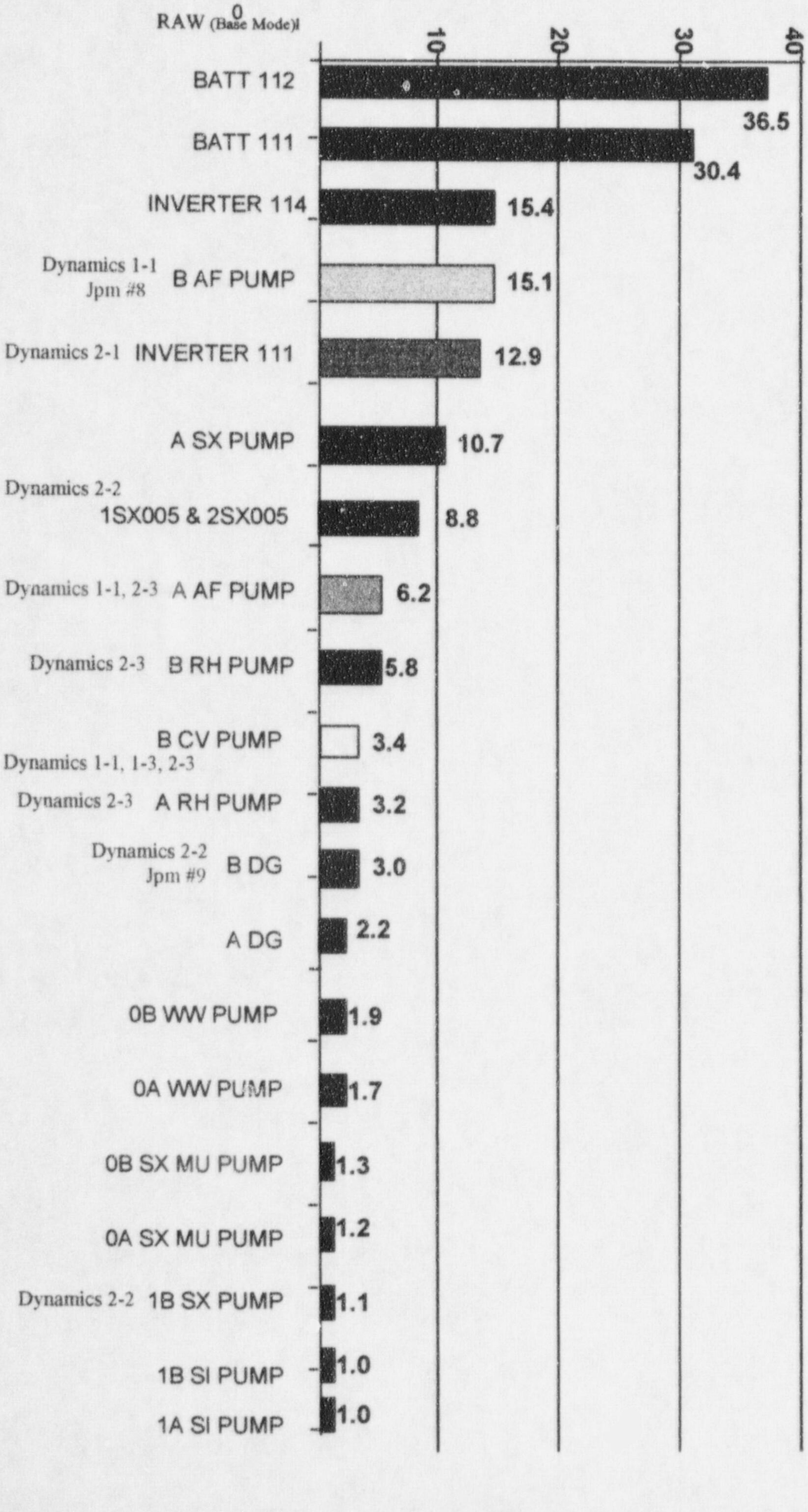
JPMs 8, 9, 10 In-plant JPMs

JPM 8 is local emergency start of 1B AFW per N-56 with modification. The initial conditions are that no feedwater flow exists to the SGs and a fire exist in such areas that control room controls and RSP controls are or may be affected. 1BOA ELEC-5 was entered for guidance on local actions of which one is to locally start the Diesel Driven Feedwater Pump. The evaluated actions are those of BOA ELEC-5 ATTACHMENT D. At step 2.d, the engine will fail to start, requiring RNO action, select alternate battery bank.

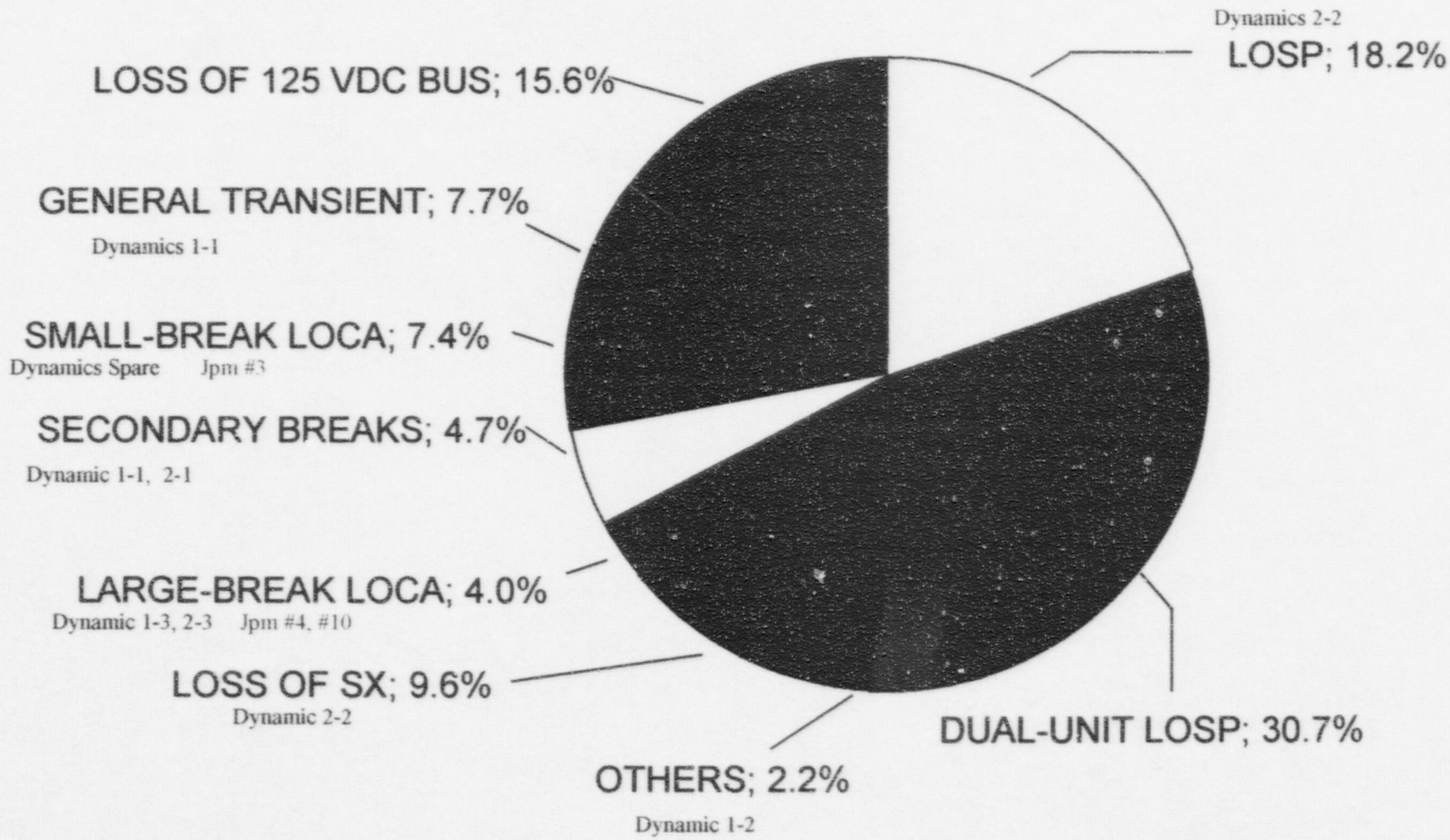
JPM 9 is abnormal start of 2A DG per N-35b. The evaluated actions of BOA ELEC-3 ATTACHMENT D steps 1-7.

JPM 10 is local alignment of CCW for Post-LOCA alignment. The initial conditions are a LOCA on Unit 1 with possible failures that could affect CCW alignment. Actions provide for train separation (U-1) with B CC Pp available. The evaluated actions are those local actions of BOP CC-14, section 2. (The actions of BOP CC-8 are completed prior to this JPM.)

Important Equipment



CDF CONTRIBUTION BY INITIATING EVENTS



OPERATING TEST NO.: 1

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

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.
(2) Reactivity manipulations must be significant as defined in Appendix D.

NOTE: Scenario Number 4 is a "spare" scenario and is represented on ES-301-5 for ALL Operating Tests for comparison purposes only in Examination Outline submittal.

The "/" in the cells for the "RO'" applicant type represents the position the applicant is expected to fill during the scenario. The events are listed for the identified position: RO / BOP.

Author: RG Brown
Chief Examiner: [signature] NOTE*

NOTE* - In scenarios the licensee was informed that (N) normal evolution - increasing Rx power is not allowed to be considered for both Reactivity change & Normal evolution. It can be counted as R or N, but not both.
ES-301, page 17 item 4.d

Item 2 - scenario 2 - can only be counted as one item (I) or (C) not both (unless it is an additional failure, then classify as an additional event (separate))

Operating Test: 1

Competencies	Applicant #1 ~~RO/~~**SRO-I**~~/SRO-U~~				Applicant #2 **RO**/~~SRO-I/SRO-U~~				Applicant #3 **RO(BOP)**/~~SRO-I/SRO-U~~			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3	4	1	2	3	4	1	2	3	4
Understand and Interpret Annunciators and Alarms	2-8	2-7	2-8	1-4, 6-7	3-4, 6-8	2-5,7	3,5-8	1-2, 4,5-7	2,5-8	3-7	2,4-8	1,3, 4,6,7
Diagnose Events and Conditions	2-8	2-7	2-8	1-4, 6-7	3-4, 6-8	2-5,7	3,5-8	1-2, 4,5-7	2,5-8	3-7	2,4-8	1,3, 4,6,7
Understand Plant and System Response	1-8	1-7	1-8	1-7	1-4, 6-8	2-5,7	1-3, 5-8	1-7	1-2, 5-8	1,3-7	1-2, 4-8	1-7
Comply With and Use Procedures (1)	1-8	1-7	1-8	1-7	1,3,4 6-8	2-5,7	1-3, 4-7	1-7	1-2, 5-8	1,3-7	1-2, 4-8	1-7
Operate Control Boards (2)	1-8	1-7	1-8	1-7	1,3,4 6-8	2-5,7	1-3, 5-8	1-7	1-2, 5-8	1,3-7	1-2, 3-5, 7-8	1-7
Communicate and Interact With the Crew	1-8	1-7	1-8	1-7	1-8	2-5,7	1-3, 5-8	1-7	1-8	1,3-7	1-2, 3-5, 7-8	1-7
Demonstrate Supervisory Ability (3)	1-8	1-7	1-8	1-7	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Comply With and Use Tech. Specs. (3)	2-4	1,2,4	3	2	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Notes:

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

Instructions:

Circle the applicant's license type and enter the event numbers that test the competency for each scenario in the set.

NOTE: **OPERATING TEST NO.: 1.** Scenario Number 4 is a "spare" scenario and is represented on ES-301-5 for ALL Operating Tests for comparison purposes only in Examination Outline submittal. The order of listing for candidates is SRO, RO and BOP by position.

Author: RG Brown
Chief Examiner: [illegible signature]

OPERATING TEST NO.: 2

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

Instructions:
(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.
(2) Reactivity manipulations must be significant as defined in Appendix D.

NOTE: Scenario Number 4 is a "spare" scenario and is represented on ES-301-5 for ALL Operating Tests for comparison purposes only in Examination Outline submittal.

The "/" in the cells for the "RO" applicant type represents the position the applicant is expected to fill during the scenario. The events are listed for the identified position: RO / BOP.

Author: RG Brown
Chief Examiner: [illegible signature]

Operating Test: 2

Competencies	Applicant #1 ~~RO~~/**SRO**~~-I/SRO-U~~				Applicant #2 **RO**/~~SRO-I/SRO-U~~				Applicant #3 **RO(BOP)**/~~SRO-I/SRO-U~~			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3	4	1	2	3	4	1	2	3	4
Understand and Interpret Annunciators and Alarms	2-8	3-8	1,2, 4-7	1-4, 6-7	2,3 6-8	3-7	1,2, 4,7	1-2, 4,5-7	2, 4-7	4-8	2,5-7	1,3, 4,6,7
Diagnose Events and Conditions	2-8	3-8	1,2, 4-7	1-4, 6-7	2,3 6-8	3-7	1,2, 4,7	1-2, 4,5-7	2, 4-7	4-8	2,5-7	1,3, 4,6,7
Understand Plant and System Response	1-8	1-8	1-7	1-7	1-3 6-8	1-7	1-4, 7	1-7	1,2, 4-7	1,2, 4-8	1-3, 5-7	1-7
Comply With and Use Procedures (1)	1-8	1-8	1-7	1-7	1-3 6-8	1,3-7	1-4, 7	1-7	1,2, 4-7	1,2, 4-8	1-3, 5-7	1-7
Operate Control Boards (2)	1-8	1-8	1-7	1-7	1-3 6-8	1,3-7	1-4, 7	1-7	1,2, 4-7	1,2, 4-8	1-3, 5-7	1-7
Communicate and Interact With the Crew	1-8	1-8	1-7	1-7	1-3 6-8	1-7	1-4, 7	1-7	1,2, 4-7	1,2, 4-8	1-3, 5-7	1-7
Demonstrate Supervisory Ability (3)	1-8	1-8	1-7	1-7	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Comply With and Use Tech. Specs. (3)	2,4	3-6	1,2	2	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Notes:

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

Instructions:

Circle the applicant's license type and enter the event numbers that test the competency for each scenario in the set.

NOTE: **OPERATING TEST NO.: 2.** Scenario Number 4 is a "spare" scenario and is represented on ES-301-5 for ALL Operating Tests for comparison purposes only in Examination Outline submittal. The order of listing for candidates is SRO, RO and BOP by position.

Author: RG Brown

Chief Examiner: [illegible]

Simulation Facility	Byron 1 & 2	Scenario No.: 1	Op Test No.: 1
Examiners: ____________		Operators: ____________	SRO
____________		____________	RO
____________		____________	BOP

Objectives: In accordance with plant procedures:

1. raise reactor power.
2. respond to trip of Component Cooling water Pump with failure of auto-start of standby pump due to header pressure instrument failure.
3. respond to failure of controlling pressurizer level channel
4. respond to degraded operation of Charging Pump.
5. respond to failure of the steam header pressure transmitter supplying feedwater pump speed control.
6. respond to a faulted steam generator with failure of ESF signal for Steamline Isolation.
7. perform actions for a loss of heat sink.

Initial Conditions:

1. IC-18, Reactor Power 75%, MOL, BGP 100-3, step F.61.
2. PZR level control is selected to 459/460.
3. 1B CV Pump running; 1A CV Pump in standby
4. SG 1B level LT-557 is out of service.
5. 1B AF pump OOS, 1A HDP OOS.

Turnover:

1. Reactor power is 75%.
2. Raise reactor power at 5 MW/hr after shift turnover.
3. 1B CV Pump is currently in service following pump seal replacement testing. 1A CV Pump is in standby.
4. Narrow range level transmitter LT-557 on SG B is out of service. All required actions have been performed per BOA INST-2 and the AR has been initiated.
5. The Diesel Driven Aux Feedwater Pump is OOS for replacement of a fuel injector. The pump has been OOS for 6 hours and is expected to be returned to service by the end of the shift.
6. 1A Heater Drain Pump is out of service for motor bearing replacement.
7. A thunderstorm warning is in effect for Stephenson, Winnebago and Ogle counties for the next 4 hours.

Event No.	Malf. No.	Event Type*		Event Description
Preload	FW43 FW44			1A and 1B AFW Pumps fail to start
Preload	RX06H, 0 MRF - RX125 trip RX072 trip RX113 trip RX146 trip			Narrow Range Steam Generator level LT-557 out of service
Preload	RP-34, OUT RP-35, OUT RP-60, OUT RP-61, OUT			Failure of MSIV automatic closure signal
Preload	MS01A, 100%			1A MSIV Fails to Close
1		R N	SRO RO BOP	Raise reactor power (5 MW/min)
2	CC01B CC02B, 144	C	BOP SRO	1A CC Pump trips coincident with 1B CC Pump discharge pressure switch failing as-is.
3	RX13A, 0, 30	I	RO SRO	Controlling pressurizer level (1LT-459) fails downscale on a 30 second ramp.
4	CV29B,50%	C	RO SRO	Running charging pump 1B experiences impeller degradation of 50% over a 5 minute period causing reduction in flow and pressure
5	RX05, 0	I	BOP SRO	Steam Header Pressure Detector PT-507 fails low
6	MS09, 3.5 MLBH	M	RO SRO BOP	Main Steam Header Break with failure of MSIV auto closure
7	(MS01A)	C	RO SRO BOP	1A Steam Generator MSIV failure to close both automatic and manual
8	(FW43)	M	RO SRO BOP	1A MD AFW pump trip results in loss of feed.

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

Peters
NOTE: suggested item 8 - that '1A' MD AFW fails to start in Auto - which requires operator to start in manual. Due to '1B' also failing to start - could have B still fail & A starts, but eventually trip and have no flow indications

SCENARIO #1-1

SCENARIO SUMMARY:

Reactor power is 75%. The 1B AF pump is OOS due to injection pump replacement, and has been OOS for the past 6 hours. The AF pump is expected to be returned to service in the next 12 hours. SG 1B Narrow Range Steam Generator Level Transmitter LT-557 is out of service. All required actions have been performed and an AR has been initiated. 1A Heater Drain Pump is out of service for motor bearing replacement. Power is to be increased after shift turnover at 5MW/min.

After power has been increased approximately 5%, the 1A CC Pump will trip and the standby 1B CC Pump will NOT auto start due to failure of the associated discharge pressure instrument. BOA PRI-6 will be entered and the operator will be required to recognize failure of auto start of the standby pump, diagnose failed pressure instrument and start the 1B CC Pump. The SRO will address Technical Specification, Condition B.

Two minutes after the actions are determined for the inoperable CC pump, the controlling channel of pressurizer level (LT-459) will fail downscale on a 30 second ramp. BOA INST-2, ATTACHMENT C will be entered to restore pressurizer level and select an operable channel. Letdown will be placed back in service and heaters restored. Channel LT-459 will be taken OOS and Technical Specification 3.3.1 Table 3.3.1-1 Function 9 actions (Condition J) addressed.

Following Technical Specification actions, the 1B Charging Pump experiences impeller degradation (to 50%) resulting in reduction of charging flow and discharge pressure. Low charging line flow alarm actuates and pump amps will decrease. Operator response should include placing the 1A Charging Pump inservice and stopping the 1B Pump. The SRO addresses Technical Specifications 3.5.2 (Condition A) and 3.5.5 (Condition A) and TRM 3.1.d.

After addressing the failed Charging Pump actions, steam header pressure transmitter PT-507 will fail downscale causing feedwater pumps to go to minimum flow. Manual control will have to be taken of the feedpump speed controller to restore normal feed flow (FW header pressure).

A Main Steam header break occurs at the steam header crosstie with a failure of the Steamline Isolation automatic closure signal and a failure of the 1A MSIV to close. The 1A AFW pump will trip on overcurrent. Feed flow is lost to the SGs. FR-H.1 will be entered and an alternate method of feedwater will be required. With 1A steam generator being faulted, the actions of BEP-2 will be required following transition from FR-H.1. The scenario ends following transition to EP-1 after isolating the fauted SG in EP-2.

ERG-Based Critical Tasks

1. EP-2 – A: Isolate the faulted SG before transition out of EP-2.
2. FR-H.1 – A: Establish feedwater flow into at least one SG before RCS bleed and feed is required.

Simulation Facility: Byron 1 & 2	Scenario No.: 2		Op Test No.: 1
Examiners: ____________________		Operators: ____________________	SRO
____________________		____________________	RO
____________________		____________________	BOP

Objectives: In accordance with plant procedures:

1. reduce reactor power (for shutdown).
2. respond to Pressurizer Pressure Master controller failure with a failure of the PORV to close.
3. respond to a loss of instrument air to the containment.
4. respond to a steam generator tube leak.
5. respond to a loss of Control Room dP.
6. perform emergency actions for a SGTR with out Pressurizer Pressure control.

Initial Conditions:

1. IC-21, Reactor Power 100%, Steady state BOL, BGP 100-3, step F.61.
2. PZR pressure control selected to 455/456.
3. Pressurizer pressure channel PT-458 is OOS with bistables tripped.
4. PORV PCV-456 control is in manual control (CLOSE) and Block Valve RY-8000B is closed and deenergized.
5. 1A HDP OOS; 250' Meteorological Tower OOS.

Turnover:

1. Reactor Power is 100%, Steady state power at BOL. BGP 100-3 is in effect.
2. Unit 2 is at 100% power.
3. The block valve (RY-8000B) for PCV-456 is closed and deenergized. A leak had developed on PCV-456. When the block valve was shut it tripped after the closed indication was observed. Electrical Maintenance is investigating. Valve has been OOS for 62 hours.
4. Pressurizer pressure instrument 1PT-458 is failed. All actions of BOA INST-2, ATTACHMENT B have been completed. PT-14 for channel has been initiated. Currently no report of expected time for release to service.
5. Heater Drain Pump 1A is out of service to meggar motor leads.
6. The 250 foot Meteorological Tower is out of service.
7. High wind warnings have been issued for Stephenson, Winnebago and Ogle counties for the next 2 hours.

Event No.	Malf. No.	Event Type*		Event Description
Preload	RF FW-78 RF FW-79 RF FW-150 RF FW-151, REMOVED	C	BOP	FWI valves fail to close (FWI signal failure)
Preload	OR ZAO0UREM002P2, 0 OR ZAO0UREM012P1, 0 OR ZAO0UREM012P2, 0			250' Met Tower OOS
Preload	RX22B, 2500 RF RX044 trip RF RX045 trip RF RX046 trip RF RX025 trip RF RX141 trip			Pressurizer pressure instrument 458 is OOS bistables tripped
Preload	RF ED065D Open			Block Valve MOV-RC8000B (PCV 456) was closed to cycle the valve. When the operators attempted to open valve, the motor operator tripped on overload.
1		R N	SRO RO BOP	Reactor shutdown due to inoperable PORV
2	RX15, 2500 TH11A, 25%	I C	RO SRO	Pressurizer Pressure Master Controller failure to maximum output/ PORV 455A fails to close requiring the block valve RY8000A to be closed.
3	IA03, 5000	C	RO BOP SRO	Loss of instrument air to the containment.
4	TH03D, 25	C	RO SRO BOP	Steam Generator 1D Tube Leak – 25 gpm.
5	TH03D, 500	M	RO BOP SRO	Steam Generator 1D Tube Rupture – Increases requiring a reactor trip and SI. FWI signal fails requiring manual operation of FW components.
6	OR ZAO0PDIVC038 0	I	BOP SRO	Main Control Room differential pressure failure with decreasing pressure.
7	RF ED058C Open	C	RO BOP SRO	Power is lost to the power supply for the Block Valve MOV-RC8000A (PORV 455A) while valve is closed (control switch taken to OPEN).

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

SCENARIO #1-2

SCENARIO SUMMARY:

The scenario will begin at 100% power. Pressure transmitter 1PT-458 is out of service. An AR has been initiated and all required actions have been taken per BOA INST-2. The block valve for PORV 456 is closed and deenergized due to thermal overloads tripping on the valve operator. The 250' elevation Meteorological Tower Instrumentation has failed and action of TRM 3.3.c is being tracked.

Operations Management will call and report that the PORV and block valve cannot be repaired within required time (16 hours from now). This will require a Unit shutdown be commenced immediately.

Following the initial down power (>5%), a failure of the Pressurizer Pressure Controller will cause PORV PCV-455A and the spray valves to open. PORV 455A fails open since the PORV enabling pressure channel PT-458 is failed HIGH. Failure of PORV 455A to close will require closure of its block valve MOV-RY8000A. The spray valves will close after manual control is taken of the pressurizer pressure controller. SRO will address Technical Specification 3.4.11 (Condition E) and TRM 3.3.k and 3.4.d.

After pressurizer pressure is returned to the normal band using manual control, instrument air will be lost to the containment. Actions will be performed in accordance with the BOA SEC-4 for loss of instrument air to the containment. The loss of instrument air will prevent operation of the pressurizer normal spray valves or the auxiliary spray valve.

A tube leak will occur in SG 1D. The crew will have to diagnose the tube leak and perform actions per the BOA PRI-1 and BOA SEC-8. This will include determination of leak size (~ 25 gpm) and SRO will address Technical Specification 3.4.13 (Condition A). A trip of the unit should be initiated due to complications of loss of pressure control with a leak.

The reactor trip will cause Steam Generator Tube Rupture to increase requiring a SI. EOP actions of BEP-0 and BEP-3 will be performed. In BEP-0, the feedwater system will not isolate due to a failure of the feedwater isolation signal requiring manual action to isolate feedwater, and at step 21, the operator will recognize failure to maintain positive Control Room pressure and will perform actions of BOP VC-14. PORV Block valve (RY8000A) electrical power feed will trip if the valve is taken to open. Without pressurizer PORVs, pressurizer normal spray valves and Aux. Spray valves, pressurizer pressure control is lost. This will require that BCA-3.3, "SGTR Without Pressurizer Pressure Control", be performed during actions for the SGTR. The scenario ends with the establishment of RCS cooldown either in BCA-3.3 or BCA-3.1, as appropriate.

ERG Based Critical Tasks:

1. EP-3 – A: Isolate feedwater flow into and steam flow from the ruptured SG before a transition to ECA-3.1 occurs.

2. EP-3 – B: Establish/maintain and RCS temperature so that transition from EP-3 does not occur because RCS temperature is in either of the following conditions.
   - Too high to maintain minimum required subcooling
     OR
   - Below the RCS temperature that causes an extreme (red-path) or a severe (orange path) challenge to the subcriticality and/or integrity CSF.

3. ECA-3.3-A Terminate SI before a water release occurs through the SG PORV or SG Safeties.

Simulation Facility Byron 1 & 2 Scenario No.: 3 Op Test No.: 1

Examiners:	____________________	Operators: ____________________	SRO
	____________________	____________________	RO
	____________________	____________________	BOP

Objectives: In accordance with plant procedures:

1. increase reactor power.
2. respond to a trip of one feed pump.
3. respond to a failed primary RTD
4. respond to failure of digital rod position indication channel.
5. respond to failed steam flow channel.
6. respond to TWO dropped control rods with failure of the reactor to trip.
7. respond to failure of running Charging Pump.
8. perform actions for a large break LOCA where containment phase B actuation fails to operate.

Initial Conditions:

1. IC-190, Reactor Power 67%, MOL, BGP 100-3, step F.61.
2. 1B CV Pump running; 1A CV Pump in standby.
3. 1A Containment Spray pump OOS.
4. 1A Motor driven Feedwater Pump OOS
5. 250' Meteorological Tower OOS.

Turnover:

1. Reactor power is 67% with power increase to continue following turnover at 5MW/hr.
2. Unit 2 is operating at 100% power.
3. 1B CV Pump is currently in service following pump seal replacement testing. 1A CV Pump is in standby.
4. 1A Containment Spray pump is 00S due to high vibrations during testing. An AR has been initiated and the pump is expected to be release to service in 24 hours.
5. 1A Feedwater Pump is OOS with electrical supply cleared due to electrical wiring problem. Pump will NOT be ready to return to service this week.
6. The 250 foot Meteorological Tower is out of service.
7. Thunderstorm warning is in effect for the next 6 hours.

Event No.	Malf. No.	Event Type*		Event Description
Preload	RP01	C	RO SRO	Reactor protection system failure of automatic trip
Preload	CS01A			1A Containment Spray pump OOS.
Preload	RP15A MRF RP83 open	C	RO SRO	Failure of Auto Start for the 1A CV pump.
Preload	OR ZDI1CS01PB, TRIP			Failure of CS Pump to start and Phase B to actuate on High-High Containment Pressure
1		R N	SRO RO BOP	Raise reactor power (5 MW/min)
2	FW02A	C	BOP SRO	1B (turbine driven) Main FW Pump trips.
3	RX18A, 630 RD13AK10	I C	RO SRO	Tcold RTD fails high resulting in higher Tavg input. Coincident with rod motion, a DATA A failure occurs for rod K10.
4	RX03B, 4.8 MLBH	I	BOP SRO	Steam Flow Transmitter FT-513 (input to controlling SGWLC) fails upscale
5	RD02K10 RD02F06 (RP01)	M	RO BOP SRO	TWO dropped control rod – Failure of the reactor to trip on negative rate trip.
6	CV01B	C	RO SRO	Running Charging Pump trip (to occur on reactor trip).
7	TH06, 50,000 to 400,000	M	RO SRO BOP	Large break LOCA inside containment. (Put on a ramp for containment pressure to rise above HI HI setpoint.)
8	(OR)	C	RO BOP SRO	Failure of Phase B and containment spray system to actuate automatically.

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

SCENARIO #1-3

SCENARIO SUMMARY:

The scenario begins with power at 67% with the 1A Containment Spray pump out of service due to high vibration during the last run. The 250' elevation Meteorological Tower Instrumentation has failed and action of TRM 3.3.c is being tracked. Power is to be raised at 5 MW/hr.

After reactor power is increased at least 5%, the 1B Main Feedwater will trip. The crew should respond per BOA SEC-1 and reduce power to within the capacity of one feedpump (540 to 600 MWe or ~ 60% power).

Prior to power stabilization following the FW Pump trip, a loop Tcold RTD will fail high resulting in increased auctioneered high Tave. This will result in demand inward motion of the control rods. The crew will perform actions of BOA ROD-1 due to the rod motion, and will be directed to BOA INST-2, ATTACHMENT A. Coincident with the RTD failure, a DATA A failure (DRPI coil fails open) will occur on CBD rod K10. Alarm Response procedures and/or BOA ROD-3 will be entered. SRO will review Technical Specification 3.3.1, Table 3.3.1-1, Function 6 & 7 (Condition D) and 3.1.7 (Condition A).

Following completion of actions for taking the failed RTD instrument out of service, the 1A SG selected flow channel instrument fails high. This results in indication of increased steam flow and initial opening of the 1A SG Feed Reg Valve to attempt to match feed flow to steam flow. An equilibrium level should be reached if manual control is not taken expediently. The operator will perform the actions of BOA INST-2, Attachment H.

Following stabilization of SG levels after control is returned to auto, TWO rods trip into the core. The reactor fails to automatically trip on the PR Negative Rate trip and the operator will have to manually trip the reactor (also required on drop of more than one rod). The running Charging Pump trips coincident with the reactor trip, and the operator must start the 1A CV Pump to provide charging flow. After performing the first FOUR steps of BEP-0, transition will be made to BEP ES-0.1.

After Step 5 is performed in BEP ES-0.1, a large break LOCA will occur requiring BEP-0 to be re-entered and actions of BEP-0 and BEP-1 to be performed. Phase B will fail to occur and containment spray will fail to initiate (i.e., the CS pumps will NOT start) requiring the crew to manually actuate CS and Phase B Isolation. The scenario ends after transition to cold leg recirculation.

ERG-Based Critical Tasks:

1. EP-0 – A: Manually trip the reactor from the Control Room before transition to ES-0.1.

2. EP-0 – E: Manually actuate at least the minimum required complement of containment cooling equipment before an extreme (red path) challenge develops to the containment CSF.

3. EP-0 – I: Manually start 1A CV pump before transition out of EP-0.

Simulation Facility	Byron 1 & 2	Scenario No.:	1	Op Test No.:	2

Examiners: ____________ ________ Operators: ____________ SRO

____________ ____________ RO

____________ ____________ BOP

Objectives: In accordance with plant procedures:

1. lower reactor power.
2. respond to 120 VAC instrument bus loss of power.
3. respond to Tref programmer failure low.
4. respond to a steam generator atmospheric dump valve failing to mid position.
5. respond to main turbine bearing high vibration.
6. perform actions for depressurization of all steam generators compounded with failure of automatic feedwater isolation to one SG.
7. respond to trip of Charging/HHSI pump with failure of the other pump to start on the SI sequence.
8. perform actions for isolation of faulted SGs

Initial Conditions:

1. IC-21, Reactor power is 100%, BOL in BGP 100-3, step F.61.
2. 1B CV Pump running; 1A CV Pump in standby.

Turnover:

1. Reactor power is 100%, Steady state power at BOL. BGP 100-3 is in effect. Power is to be reduced to allow performance of TV/GV surveillance.
2. Unit 2 is operating at 100% power.
3. 1B CV Pump is currently in service checking for post-test leakage. 1A CV Pump is in standby.
4. Weather conditions are high wind warnings for Stephenson, Winnebago and Ogle counties for the next 6 hours.
5. Night orders indicate that security has received threats of sabotage against ComEd nuclear plants.

Event No.	Malf. No.	Event Type*		Event Description
Preload	RF FW-150, RF FW-151, REMOVED	C	BOP	Failure of feedwater Reg valve to close on FWI for SG 1C. Failure of Feedwater Isolation Valve to close for SG 1C.
Preload	RP15A RF RP83, OPEN	C	RO	1A CV Pump fails to sequence on SI signal. Manual start available.
Preload	MS01A, 100% MS01B, 100% MS01C, 100% MS01D, 100%	C	RO BOP SRO	MSIVs fail to close
1		R N	RO BOP SRO	Lower reactor power.
2	ED11A	C	RO BOP SRO	120 VAC Instrument Bus 111 inverter failure.
3	RX12, 557	I	RO SRO	Tref programmer output fails low.
4	MS04A, 50%	I	BOP SRO	Steam Pressure Instrument 1PT-MS041 fails high failing Atmospheric Dump valve to mid position on SG 1A.
5	TU01I, 15, 600 sec	C	BOP SRO	Turbine bearing #9 increasing vibration over 5 minutes with turbine trip required.
6		M	RO BOP SRO	Turbine Trip due to High Vibrations
7	MS08A, 4 MLBH (RF FW-150)	M	RO BOP SRO	Main steam rupture downstream of the MSIV on 1A SG with a failure of the MSIVs to close. The crew will be able to locally close valves on two loops. **NOTE:** MS01 for two SGs (1B & 1C) should be removed sequentially following dispatch of operators to locally close valves. Operator should close first two valves in order if sequence provided by crew. SG FW fails to isolate
8	CV01B (RP15A)	C	RO	On the SI signal, the running Charging Pump trips. NO CV pump starts on SI sequence.

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

SCENARIO #2-1

SCENARIO SUMMARY:

The scenario will begin at approximately 100% power. After shift turnover the crew will lower load in order to perform main turbine Throttle Valve/Governor Valve surveillance testing.

Following at least a 5% reactor power change, the inverter to Vital Instrument Bus 111 fails causing a loss of power to the bus. The actions of BOA ELEC-2, ATTACHMENT A will be performed and bus re-energized from CVT. The SRO will address Technical Specification 3.8.7 (Condition A) and 3.8.9 (Condition B). (See next page for major affected components.)

When power is restored to Bus 111, the output of the Tref programmer will fail to the low value of 557°F. This will affect rod control causing rods to step in on Tave-Tref error. Actions of BOA ROD-1 may be performed due to the rod motion. The crew will restore Tave to actual Tref (BCB-1 Figure 33 or equivalent may be used).

After power is stabilized following the Tref failure, steam pressure transmitter PT-041 (or PORV controller) on SG 1A will fail high causing MS PORV MS018A (atmospheric relief valve) to open. When manual control is taken of the atmospheric dump valve it will only partially close (50% open). The crew will have to locally isolate the atmospheric dump valve (MS019C). SRO will address Technical Specification 3.7.4 (Condition A).

Following isolation of the SG 1C MS PORV, a vibration for the turbine #9 bearing will increase over a period of time. The crew will perform actions of BOA TG-1. The reactor will be manually tripped and the turbine will be tripped as vibration continues to rise toward the trip setpoint. On the trip the crew will enter BEP-0.

When the turbine is tripped a steam rupture will occur on the steam header. All MSIVs will fail to close in response to the steam leak and efforts to close the MSIVs from the control room will be unsuccessful. Additionally the feedwater isolation signal for the steam generator will fail. On the SI signal, the running Charging/HHSI Pump will trip and the 1A CV pump will not auto start when the SI sequencing occurs. The crew will manually start the CV pump. Transition is made ECA-2.1 due to all SGs depressurizing. After step 8 of ECA-2.1 the local efforts to close two of the MSIVs (preferably 1B & 1C SGs) will be successful. The reports of success will be staggered. The scenario ends following transition to BEP-2, identification and attempted isolation of remaining faulted SGs and decision to transition to BEP-1.

ERG Based Critical Tasks

1. EP-0 – I: Establish flow from at least one high-head ECCS pump before transition out of E-0.
2. EP-0 – P: Manually actuate main steamline isolation 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.
3. ECA-2.1 – A: Control the AFW 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 integrity CSF.
4. EP-2 – A: Isolate the faulted SG before transition out of EP-2.

Simulation Facility Byron 1 & 2 Scenario No.: 2 Op Test No.: 2

Examiners: ______________________ Operators: ______________________ SRO

______________________ ______________________ RO

______________________ ______________________ BOP

Objectives: In accordance with plant procedures:

1. reduce reactor power.
2. transfer to Feed Reg Bypass valves.
3. respond to a pressurizer pressure instrument failing high.
4. recognize and perform required action for P-7 failing to perform as required.
5. respond to a service water pump trip..
6. respond to 4160V ESF bus lockout.
7. perform required actions for a loss of off-site power including restoring power to one ESF bus.
8. Respond to a failure of MSIV's to close from a Main Steam Line Isolation Signal

Initial Conditions:

1. IC-191, Reactor power 21%.
2. BGP 100-4 is complete through step F.16.
3. Pzr pressure control is selected to 455/456
4. 1B DG OOS
5. 1B SX Pump is running for Unit due to 2B SX pump OOS.

Turnover:

1. Reactor power is 17% during a reactor startup.
2. BGP 100-4 is complete through step F.16.
3. Continue unit shutdown and transfer feedwater control to the Feed Reg Bypass Valves.
4. The 1B Diesel Generator is inoperable due to failure to meet time requirements for rated voltage and speed. Maintenance is investigating governor for problem. All required actions have been completed on prior shift. Next surveillance (SR 3.8.1.1) due in four hours.
5. The 1B SX pump is running. The 2B SX Pump is out of service due to seal leak, and so the 2A SX Pump is in service.
6. Unit 2 is shutdown.

Event No.	Malf. No.	Event Type*		Event Description
Preload	RP17A & B	I		P-7 fails as is.
Preload	IOR ZDIMSI1 Normal ZDIMS2 Normal	C	BOP SRO	Failure of Main Steam line Actuation Signal.
1		R N	RO BOP SRO	Load decrease from 21%.
2		N	BOP SRO	Transfer the Feedwater Control from the Main Feed Reg Valves to the Feed Reg Bypass Valves.
3	RX21A,2500	I	RO SRO	Pressurizer pressure instrument 1PT-455 fails high.
4	RX10A, 800 (RP17A & B)	I I	RO BOP SRO	PT-505 turbine first stage pressure fails high in conjunction with P-7 at ~10% power. At-power Trips fail to be clear when power is lowered below 10%.
5	SW01B	C	SRO BOP	Service water pump 1B trips.
6	ED07A	C	RO SRO BOP	Lockout on ESF Bus 141.
7	ED15D	M	RO SRO BOP	Loss of off-site power. Fault on Bus 6. Reactor trip required due to loss of SX. Power will be available by electrical crosstie to Unit 2, Bus 242
8	IOR ZDIMSI1 Normal ZDIMS2 Normal	C	BOP SRO	Failure of Main Steam line Actuation Signal.

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

SCENARIO #2-2

SCENARIO SUMMARY:

The scenario will begin at approximately 21% power with load decrease to shutdown conditions to continue. The 1B DG is inoperable with maintenance investigating.

During load decrease the crew will transfer the Feedwater Control from the Main Feed Reg Valves to the Feed Reg Bypass Valves.

Following transfer to Bypass FRVs, pressurizer pressure instrument 1PT-455 will fail upscale causing both spray valves to open. Actions are taken per BOA INST-2, ATTACHMENT B. The crew will select an operable Pzr pressure channel for control and then attempt to close the spray valves. SRO will address Technical Specifications 3.3.1 Table 3.3.1-1 Functions 6, 8 (Conditions D & J), 3.3.2 Table 3.3.2-1 Function 1.d (Condition K).

Just prior to reaching 10% reactor power or turbine power, one channel of input to P-13, First Stage Pressure PT-505 fails high. This provides an input to P-7. Action s are taken per BOA INST-2, ATTACHMENT D. When power is reduced to < 10%, the IM will be required to defeat the P-13 input to P-7, in order to block (bypass) the "at-power" reactor trips. SRO will address TRM 3.3.z Table 3.3.z-1 Functions b & e (Condition B).

After the crew determines the required course of action for PT-505 failure, Service Water Pump 1B trips. The crew will start the 1A SX pump as directed by BOA PRI-7 and direct investigation of the tripped pump. SRO will address Technical Specification 3.7.8 (Condition A).

After the Technical Specifications are addressed for the service water pump, a lockout on ESF 141 will result in the loss of 1 division of electrical power. The crew will have to start equipment on other trains where required. Loss of this bus will contribute to the loss of all ac when off-site power is lost. 1A DG will have to be stopped since its output breaker will not close to the bus and no SX pumps will be available on the unit. SX will be crosstied to Unit 2 as directed by BOA PRI-7, ATTACHMENT A. A problem with Unit 2 cross connect valve(s) will delay completion until after initiation of the following event.

A loss of offsite power occurs when Bus section 6 fault occurs. The fault trips infeeds to SATs and results in all 4KV AC buses deenergized. DG 1A output breaker will NOT close to bus 141 due to lockout and bus 142 is deenergized. With the loss of all AC power, BCA-0.0 is implemented. A failure of Main Steam line Actuation Signal will require closing the individual MSIV from their control switches. Power to bus 142 will be restored by crosstie to Unit 2 bus 242. Scenario ends with actions with transition to recovery procedure at step 54 of BCA-0.0.

ERG Based Critical Tasks

1. EP-0 – C: Energize at least one ac emergency bus before transition out of EP-0 unless transition is to ECA-0.0, then energize bus before placing safeguards equipment in PTL.

2. ECA-0.0 – H: Isolate RCP seal injection before a charging pump starts or is started.

Simulation Facility Byron 1 & 2 Scenario No.: 3 Op Test No.: 2

Examiners: ______________________ Operators: ______________________ SRO

______________________ ______________________ RO

______________________ ______________________ BOP

Objectives: In accordance with plant procedures:

1. respond to failure of a Pressurizer Level instrument
2. respond to an RCS leak.
3. reduce reactor power
4. respond to failure of a CV valve diverting letdown from VCT.
5. respond to a failure of the condenser hotwell level controller
6. respond to trip of a Condensate/Condensate Booster Pump set.
7. perform actions for large break LOCA
8. respond to trip and failure to start for LHSI pumps.

Initial Conditions:

1. IC-16, 53% power MOL, BGP 100-3, Step F.61.
2. PZR level control is set to 459/460.
3. Remove 1A AF pump from service.
4. Remove 1A Charging Pump from service.
5. Remove 1A Motor Driven feedpump from service.

Turnover:

1. Current power is 53%, MOL. Power will be raised following final verification of calorimetric data.
2. Unit 2 is shutdown for refueling.
3. A high wind warning has been issued for Stephenson, Winnebago and Ogle counties for the next six hours.
4. 1A AF Pump is OOS due to grounds. An AR has been initiated. The expected time OOS is at least 24 hours. Tagging clearance has been hung but has not been verified.
5. 1A Charging Pump is OOS for preventative maintenance. The pump is expected to be released to service in the next 2 hours.
6. Motor Driven feedpump OOS for bearing replacement. The pump will be OOS for the entire shift.

Event No.	Malf. No.	Event Type*		Event Description
Preload	FW43			MD AFW Pump 1A OOS
Preload	CV01A			Charging pump 1A OOS
Preload	FW01			Motor Driven Feedwater pump OOS.
Preload	RH01A	C	RO	Trip of 1A RH Pump
Preload	RP15F RF RP85 open	C	RO	Failure of 1B RH Pump to auto start on SI sequence
Preload	RF FW049, 100 FW050, 100 FW052, 0	C	BOP	
1	RX13A, 0	I	RO SRO	Pressurizer level channel 1LT-459 fails downscale causing letdown isolation. Control will be transferred to another channel and letdown restored.
2	TH06D, 20	C	RO BOP SRO	RCS leak 1D RCS Cold Leg loop leak - 15 gpm.
3		R N	SRO RO BOP	Reduce power due to RCS leakage.
4	CV4, 100	C	RO SRO	1CV112A VCT Divert valve fails to DIVERT position. **NOTE:** Valve may be returned to VCT position by failing air locally. If crew directs so, Sim Operator will change fail position to 0.
5	FW37, 48" FW38, 48" IOR ZAO101CD042, 48	I	BOP SRO	Hotwell level controllers (LT-CD037, LT-CD038) fail high causing actual hotwell level to decrease.
6	FW22C	C	BOP SRO	1C CD/CB pumps trip
7	TH06D, 540000 (RP15A)	M	RO BOP SRO	Large break LOCA with failure of RHR pumps to auto start

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

SCENARIO #2-3

SCENARIO SUMMARY:

The scenario will begin at 53% power with 1A AFW Pump, 1A Charging Pump, and the Motor Driven Feedwater Pump out of service. Preparations may be made for load increase when informed of completion of calorimetric data verification

Pressurizer level channel 1LT-459 failure low will cause letdown to isolate, heaters to shutoff and associated alarms. Charging flow rate will increase causing an increase in actual PZR level. The operators will have to select an operable level instrument then restore letdown and heaters. SRO will address Technical Specification 3.3.1 Table 3.3.1-1 Function 9 (Condition J).

A small RCS leak (15 gpm) will occur on RCS D loop requiring the crew to determine the leak size, take action to maintain pressurizer level. BOA PRI-1 will be entered and SRO will address Technical Specification 3.4.13 (Condition A). The crew will initiate a unit shutdown due to the leak. (Operations Management will provide direction for shutdown if necessary.)

During the power reduction and following at least 5% reactor power change, the VCT letdown divert valve 1CV112A will fail to the DIVERT position directing letdown flow to the Hold Up Tank. This will cause a reduction in actual VCT level and automatic makeup on low level. An operator may be dispatched to locally check the valve and will report failure of solenoid to deenergize. The crew may direct isolation of air to the valve that will cause it to fail to the VCT position.

Following actions to maintain VCT level, the normal hotwell level controllers (1LC-CD037) 1LC-CD038 fails high. This results in actual hotwell level decrease and actuation of several alarms associated with opening of 1CD141, Emergency Overflow Valve, and lowering hotwell level. Isolation of 1CD141 by closing manual isolation valve will be required or swap to alternate channel.

After hotwell conditions are restored, a Condensate/Condensate Booster Pump trips. BOA SEC-1, ATTACHMENT B is entered and actions performed. The operator should start the standby CD/CB pump and check feedwater flow conditions stable.

Immediately following completion of actions for tripped CD/CB Pumps, a large break LOCA occurs resulting in SI actuation. BEP-0 actions will be performed. The 1A RH Pump will trip when it attempts to start, and the 1B RH Pump will fail to automatically start on the SI sequencing. The operator will manually start the 1B RH Pump.

The crew will transition to BEP-1 following diagnosis in BEP-0. When the RWST level falls below 46%, the crew will transition to BEP ES-1.3, and align the "B" train for recirculation mode. The scenario will end following alignment of Containment Spray system for recirculation.

ERG-Based Critical Tasks

1. EP-0 – H: Manually start at least one low-head ECCS pump before transition out of EP-0.
2. ES-1.3 – A: Transfer to cold leg recirculation and establish ECCS recirculation flow that at least meets the assumptions of the LOCA analysis (one train).

Simulation Facility	Byron 1 & 2	Scenario No.:	Spare	Op Test No.:	1/2
Examiners:	____________	Operators:	____________	SRO	
	____________		____________	RO	
	____________		____________	BOP	

Objectives: In accordance with plant procedures:

1. respond to FW Pump trip.
2. respond to the failure of a Narrow Range RTD.
3. respond to a dropped/misaligned rod.
4. respond to failure of a containment pressure channel.
5. respond to RCP seal failures.
6. reduce reactor power.
7. respond to RCS LOCA with failure of selected ECCS valves.

Initial Conditions: 1. IC-18, Reactor Power 75%, MOL, BGP 100-3, Step F.61.

Turnover: 1. Reactor power is 75%. Power is to be increased later in the shift at the direction of the BPO.

Event No.	Malf. No.	Event Type*		Event Description
Preload	RF RP-49, OUT RP-75, OUT			RWST suction valves, VCT outlet valves, HHSI injection header discharge valves, and normal charging MOVs fail to reposition on the SI actuation. (CV112B, C, D & E; SI8801A & B; CV8105 & CV8106)
Preload	FW01A			1A Motor driven Main FW Pump trips on start.
1	FW02B	C	BOP SRO	1C Turbine driven Main FW Pump trips.
2	RD03H08 RX18B, 630, 30 sec	C I	RO SRO	Primary Cold Leg Narrow Range RTD fails high (30 sec ramp), when rods begin moving in a rod in bank D will ratchet in resulting in rod position deviation.
3	CH08C, 60	I	BOP SRO	Containment pressure channel PT-CS936 fails high.
4	CV28A	C	RO BOP SRO	1A RCP #2 seal failure
5		R N	SRO RO BOP	Power reduction to remove RCP from service.
6	CV27A, 300	M	RO SRO BOP	1A RCP #1 seal failure results in LOCA condition via RCP 1A seals. **NOTE: Simulator operator will enter LOCA on 1 minute ramp**
7	(RF)	C	RO SRO BOP	Charging system valves fail to automatically transfer upon SI initiation.

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

SCENARIO #Spare

SCENARIO SUMMARY:

Reactor power is 75%. After the crew assumes the shift a main feed pump will trip. If the motor-driven feed pump start is attempted, the pump will fail to start. The crew should respond per BOA SEC-1 and reduce power to within the capacity of one feedpump (540 to 600 MWe or ~ 60% power).

After power is reduced to within the capability of the feedwater system and prior to stabilization, a Narrow Range Cold Leg RTD will fail high causing Tave to increase. This will cause control rods to move inward. When movement is initiated a control rod H8 will ratchet in due to movable gripper failure. Initial actions are taken per BOA INST-2, ATTACHMENT A for the RTD failure and the Technical Specification 3.3.1 Table 3.3.1-1 Functions 6 & 7 (Conditions D) will be addressed. Actions of BOA ROD-3 are performed for the misaligned rod and the appropriate Technical Specification 3.1.4 (Condition A & B) and 3.1.6, if applicable, are addressed.

After conditions are stabilized from the control rod mispositioning and actions are taken for the failed temperature instrument, a containment pressure channel will fail high. The crew will perform actions of BOA INST-2, ATTACHMENT J. The SRO will address Technical Specification 3.3.2, Table 3.3.2-1, Functions1.c, 2.c, 3.b.(3), 4.c (Conditions F & G).

Following completion of actions for the failed containment pressure channel, #2 seal failure for RCP 1A will occur. Actions are addressed in BOA RCP-1. Power reduction will be initiated (if required, at the direction of Operations Management). During the power reduction and after power is lowered at least 5%, a failure of Control Rod Drive circuit will prevent rod movement in automatic or manual. The crew will evaluate conditions and should determine a reactor trip is required due to failure of reactivity control system with a reactor shutdown required.

After power is lowered at least 5%, the #1 seal on RCP 1A fails resulting in a LOCA through the RCP seals. The actions of BEP-0 and BEP-1 will be performed. When a SI is required, the charging pumps will fail to automatically transfer to the RWST and the discharge valves in the HHSI injection line will fail to open, requiring the operators to manually open the valves and close valves for normal charging path. Actions will be taken for the small break LOCA that occurs on the RCP. Scenario ends with initiating of RCS cooldown, if RCS cooldown and depressurization is required.

ERG Based Critical Tasks

1. EP-0 – I: Establish flow from at least one high-head ECCS pump before transition out of EP-0.

2. EP-1 – C: Trip all RCPs so that CET temperatures do not become superheated when forced circulation in the RCS stops.

Facility: Byron 1 & 2 Date of Exam: September 14, 1998 Exam Level: SRO

Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	4	2	4				5	7			2	24
	2	2	1	3				5	3			2	16
	3			1				1	1				3
	Tier Totals	6	3	8				11	11			4	43
2. Plant Systems	1	2	1	1	1	2		1	3	2	2	4	19
	2	1	1	2	3	1	1	2	2	1	1	2	17
	3	1		1					1	1			4
	Tier Totals	4	2	4	4	3	1 *	3	6	4	3	6	40

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	6	6	1 *	4	17

Note:
- Attempt to distribute topics among all K/A categories; select at least one topic from every K/A category within each tier.
- Actual point totals must match those specified in the table.
- Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.
- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

* Noted - to licensee - may consider increasing these area - to at least 2 items - to ensure balance.

Facility: Byron Exam Date: 9/14/98 Examination Level: SRO

Section Title Generic Knowledge and Abilities

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Conduct of Operations	2.1.1	3.8	Knowledge of conduct of operations requirements.	B	Evaluation of requirement for "active" license
	2.1.1	3.8	Knowledge of conduct of operations requirements.	B	Direction of NLO personnel
	2.1.2	4.0	Knowledge of operator responsibilities during all modes of plant operation.	B	Operating Daily Orders
	2.1.13	2.9	Knowledge of facility requirements for controlling vital / controlled access.	S	US responsibility on CNMT entry
	2.1.14	3.3	Knowledge of system status criteria which require the notification of plant personnel.	B	MOV Stroke Time actions
	2.1.23	4.0	Ability to perform specific system and integrated plant procedures during all modes of plant operation.	B	Proceudre required usage
Equipment Control	2.2.12	3.4	Knowledge of surveillance procedures.	S	Equipment operability during surveillance
	2.2.13	3.8	Knowledge of tagging and clearance procedures.	B	MSIV tagout to prevent opening
	2.2.22	4.1	Knowledge of limiting conditions for operations and safety limits.	S	Technical Specification 3.0.3 application
	2.2.23	3.8	Ability to track limiting conditions for operations.	S	Timing for Tech Spec required Shutdown
	2.2.26	3.7	Knowledge of refueling administrative requirements.	B	RCS level discrepancy during refueling
	2.2.32	3.3	Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	B	RO duties in Control Room during refueling
Radiation Control	2.3.1	3.0	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	B	Radiation exposure determination
Emergency Procedures / Plan	2.4.16	4.0	Knowledge of EOP implementation hierarchy and coordination with other support procedures.	B	Performance of Status Trees/Function Restoration

Facility: Byron Exam Date: 9/14/98 Examination Level: SRO

Section Title Generic Knowledge and Abilities

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Emergency Procedures / Plan	2.4.20	4.0	Knowledge of operational implications of EOP warnings, cautions, and notes.	B	Applicability of Operator Action Summary Page
	2.4.29	4.0	Knowledge of the emergency plan.	S	Hazmat Spill Response
	2.4.31	3.4	Knowledge of annunciators alarms and indications, and use of the response instructions.	B	Identification of inoperable CR annunciators

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Plant Systems

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Control Rod Drive System	001000A206	3.7	Effects of transient xenon on reactivity	B	Effect of Xenon Transient & compensation
	001000K103	3.6	CRDM	B	Application of DC Hold
Reactor Coolant Pump System	003000A106	3.1	PZR spray flow	B	RCP and Pzr spray operations
	003000K110	3.2	RCS	S	SG temperature effect upon start of RCP
Chemical and Volume Control System	004000A407	3.7	Boration/dilution	B	Calculation of RCS Boron dilution
Engineered Safety Features Actuation System	013000A301	3.9	Input channels and logic	B	CNMT Spray/Phase B
	013000G110	3.9	Knowledge of conditions and limitations in the facility license.	S	Conditions for MODE change - ESFAS function inop
Rod Position Indication System	014000K502	3.3	RPIS independent of demand position	S	DRPI vs. Demand Position
Nuclear Instrumentation System	015000A202	3.5*	Faulty or erratic operation of detectors or compensating components	B	SR NIS discriminator failure
	015000K201	3.7	NIS channels, components, and interconnections	B	SR NIS - loss of control power
	015000K405	4.5	Reactor trip	S	Work on PR NIS affect on SR NIS
Containment Spray System	026000A208	3.7	Safe securing of containment spray when it can be done)	B	Sequence for securing CNMT Spray
	026000G110	3.9	Knowledge of conditions and limitations in the facility license.	S	CNMT Spray/Spray Additive operability requirements - Basis
Main Feedwater System	059000G107	4.4	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	B	SG Level program - low power
Auxiliary / Emergency Feedwater System	061000A301	4.2	AFW startup and flows	B	AFW Startup
	061000K502	3.6	Decay heat sources and magnitude	B	AFW flow requirements for cooldown

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Plant Systems

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
D.C. Electrical Distribution	063000G130	3.4	Ability to locate and operate components, including local controls.	B	DC bus battery charger
Liquid Radwaste System	068000A404	3.7	Automatic isolation	B	RCDT operation - effect of CNMT Isolation
Area Radiation Monitoring System	072000K302	3.5	Fuel handling operations	B	Loss of FHB Overhead Crane rad monitor

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Plant Systems

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Reactor Coolant System	002000A111	3.2	Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling	B	Relationship of levels during refueling operations
	002000G110	3.9	Knowledge of conditions and limitations in the facility license.	S	Conditions for loops operable/in operation
Emergency Core Cooling System	006000A213	4.2	inadvertent SIS actuation	B	Systems response to SI/Actions
	006000K302	4.4	Fuel	B	10CFR50.46 Design Criteria
	006000K603	3.9	Safety Injection Pumps	B	Evaluation of flow ECCS pumps
Pressurizer Pressure Control System	010000A108	3.3	Spray nozzle DT	B	Spray using Normal and Aux Spray
	010000G112	4.0	Ability to apply technical specifications for a system.	S	DNB Limits
	010000K501	4.0	Determination of condition of fluid in PZR, using steam tables	B	Evaluation of Pzr conditions
Pressurizer Level Control System	011000K104	3.9	RPS	B	Pzr Level Reactor Trip
Reactor Protection System	012000A403	3.6	Channel blocks and bypasses	B	Input that can be bypassed & plant conditions
	012000K402	4.3	Automatic reactor trip when RPS setpoints are exceeded for each RPS function; basis for each	S	Basis for OTdT with input
Non-Nuclear Instrumentation System	016000K302	3.5*	PZR LCS	B	NR RTD Failure effects
Spent Fuel Pool Cooling System	033000A201	3.5	Inadequate SDM	S	Safety Analysis on dilution during refueling
A.C. Electrical Distribution	062000K201	3.4	Major system loads	S	Evaluation of Electrical Supplies
Emergency Diesel Generators	064000A307	3.7	Load sequencing	B	Sequencing of CNMT Spray pumps - SI & SI w LOP
Process Radiation Monitoring System	073000K401	4.3	Release termination when radiation exceeds setpoint	S	Containmen Ventilation (Purge)

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Plant Systems

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Fire Protection System	086000K406	3.3	CO2	B	Effect of loss of DC - CO2 actuation

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Plant Systems

SRO Group 3

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Residual Heat Removal System	005000A202	3.7	Pressure transient protection during cold shutdown	S	Requirements/Operation of PORVs at low RCS pressures
	005000K112	3.4	Safeguard pumps	B	Recirc interties to SI Pumps & CV Pumps
Steam Dump System and Turbine Bypass Control	041000A302	3.4	RCS pressure, RCS temperature, and reactor power	B	Steam Dump input malfunction
Instrument Air System	078000K302	3.6	Systems having pneumatic valves and controls	B	Evaluation of eqpt affected for slow loss of IA

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Continuous Rod Withdrawal	000001A205	4.6	Uncontrolled rod withdrawal, from available indications	B	Evaluate conditions - unwarranted rod withdrawal
	000001K103	4.0	Relationship of reactivity and reactor power to rod movement	S	Reactivity effect with positive MTC
Dropped Control Rod	000003K310	4.2	RIL and PDIL	B	P/A vs. Group Step Counters
Inoperable/Stuck Control Rod	000005K106	3.8	Bases for power limit, for rod misalignment	S	Reason for power reduction
Large Break LOCA	000011A103	4.0	Securing of RCPs	B	RCP trip criteria evaluation
	000011A204	3.9	Significance of PZR readings	S	Pzr level requirements
	000011K312	4.6	Actions contained in EOP for emergency LOCA (large break)	S	Use of Adverse containment
Reactor Coolant Pump Malfunctions	000015A210	3.7	When to secure RCPs on loss of cooling or seal injection	B	Eval loss of cooling flow (CCW)
	000015K207	2.9	RCP seals	B	Eval of RCP seal failure
Emergency Boration	000024A205	3.9	Amount of boron to add to achieve required SDM	B	Time/amount E-boration for condition
Loss of Component Cooling Water	000026A105	3.1	The CCWS surge tank, including level control and level alarms, and radiation alarm	B	Evaluation of CCW leak
	000026G224	3.8	Ability to analyze the affect of maintenance activities on LCO status.	S	Determine Tech Spec limitation on eqpt outage
Anticipated Transient Without Scram	000029G448	3.8	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.	B	AMS conditions
Steam Line Rupture	000040A101	4.6	Manual and automatic ESFAS initiation	B	Steamline isolation
	000040K106	3.8	High-energy steam line break considerations	B	Eval of Leak
Loss of Condenser Vacuum	000051A202	4.1	Conditions requiring reactor and/or turbine trip	B	Eval of conditions
Station Blackout	000055A104	3.9	Reduction of loads on the battery	S	Evolutions required to be performed to reduce DC loads

Facility: Byron | Exam Date: 9/14/98 | Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Station Blackout	000055K302	4.6	Actions contained in EOP for loss of offsite and onsite power	B	Identification of RCP seal LOCA/cooldown
Loss of Vital AC Instrument Bus	000057A219	4.3	The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus	B	Eqpt affected on bus loss
Control Room Evacuation	000068A121	4.1	Transfer of controls from control room to shutdown panel or local control	B	Operations required for transfer
Inadequate Core Cooling	000074K103	4.9	Processes for removing decay heat from the core	B	Major action categories
High Reactor Coolant Activity	000076A202	3.4	Corrective actions required for high fission product activity in RCS	B	Actions for reducing activity
Pressurized Thermal Shock	00WE08K202	4.0	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	B	Identification of heat removal process
Natural Circulation Operations	00WE09K301	3.6	Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.	B	Natural Circ conditions and limits

Facility: Byron Exam Date: 9/14/98 Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Reactor Trip	000007A103	4.1	RCS pressure and temperature	B	Stabilized RCS temperature with failure of Steam Dumps
	000007A201	4.3	Decreasing power level, from available indications	S	EP-0 indications
Pressurizer Vapor Space Accident	000008G222	4.1	Knowledge of limiting conditions for operations and safety limits.	S	Evaluation of PORV leak - Tech Spec Limits
Small Break LOCA	000009A110	3.9*	Safety parameter display system	B	Calculation of subcooled margin on Iconic Display
	000009K316	4.1	Containment temperature, pressure, humidity and level limits	S	Effects of Adverse containment
Loss of Reactor Coolant Makeup	000022A108	3.3	VCT level	B	VCT level transmitter malfunction
Loss of Residual Heat Removal System	000025K101	4.3	Loss of RHRS during all modes of operation	B	Calc of time to saturation/core boiling
	000025K301	3.4	Shift to alternate flowpath	B	Alternate RCS cooling
Pressurizer Pressure Control Malfunction	000027A101	3.9	PZR heaters, sprays, and PORVs	B	Pressure controller step change
	000027A215	4.0	Actions to be taken if PZR pressure instrument fails high	B	Non-Controlling channel failure
Loss of Source Range Nuclear Instrumentation	000032K101	3.1	Effects of voltage changes on performance	B	Evaluation of SR NIS voltage failure
Loss of Intermediate Range Nuclear Instrumentation	000033A204	3.6	Satisfactory overlap between source-range, intermediate-range and power-range instrumentation	B	Eval of failed IR channel on SU
Steam Generator Tube Leak	000037G132	3.8	Ability to explain and apply all system limits and precautions.	S	SG tube leak increase actions
Steam Generator Tube Rupture	000038K306	4.5	Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures	B	Loss of subcooling

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Loss of Secondary Heat Sink	00WE05K201	3.9	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	B	Interlocks affecting reestablishment of feed
Loss of Emergency Coolant Recirculation	00WE11A101	4.0	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	B	Reason for rapid SG depressurization

Facility: Byron Exam Date: 9/14/98 Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 3

System/Evolution	K/A	SRO	KA Statement	Level	Question Topic
Pressurizer Level Control Malfunction	000028K305	4.1	Actions contained in EOP for PZR level malfunction	B	Failed level channel low.
Loss of Off-Site Power	000056A121	3.3*	Reset of the ESF load sequencers	B	Reset of sequencer
	000056A246	4.4	That the ED/Gs have started automatically and that the bus tie breakers are closed	B	Eval of electric bus status

Facility: Byron 1 & 2 Date of Exam: September 14, 1998 Exam Level: RO

Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	2	2	2				4	6				16
	2	3	2	3				6	2			1	17
	3			1				1	1				3
	Tier Totals	5	4	6				11	9			1	36
2. Plant Systems	1	3	2	1	2	2	1	1	2	3	4	2	23
	2	2		2	2	2	1	2	2	3	2	2	20
	3	2		1	1				1	1	1	1	8
	Tier Totals	7	2	4	5	4	2	3	5	7	7	5	51

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	5	3	2	3	13

Note:
- Attempt to distribute topics among all K/A categories; select at least one topic from every K/A category within each tier.
- Actual point totals must match those specified in the table.
- Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.
- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Generic Knowledge and Abilities

RO Group 1

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Conduct of Operations	2.1.1	3.7	Knowledge of conduct of operations requirements.	B	Evaluation of requirement for "active" license
	2.1.1	3.7	Knowledge of conduct of operations requirements.	B	Direction of NLO personnel
	2.1.2	3.0	Knowledge of operator responsibilities during all modes of plant operation.	B	Operating Daily Orders
	2.1.14	2.5	Knowledge of system status criteria which require the notification of plant personnel.	B	MOV Stroke Time actions
	2.1.23	3.9	Ability to perform specific system and integrated plant procedures during all modes of plant operation.	B	Procedure required usage
Equipment Control	2.2.13	3.6	Knowledge of tagging and clearance procedures.	B	MSIV OOS hang to prevent opening
	2.2.26	2.5	Knowledge of refueling administrative requirements.	B	RCS level discrepancy during refueling
	2.2.32	3.5	Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	B	RO duties in Control Room during refueling
Radiation Control	2.3.1	2.6	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	B	Radiation exposure determination
	2.3.10	2.9	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	R	Fuel Handling Accident Response
Emergency Procedures / Plan	2.4.16	3.0	Knowledge of EOP implementation hierarchy and coordination with other support procedures.	B	Performance of Status Trees/Function Restoration
	2.4.20	3.3	Knowledge of operational implications of EOP warnings, cautions, and notes.	B	Applicability of EOP Operator Action Summary Page
	2.4.31	3.3	Knowledge of annunciators alarms and indications, and use of the response instructions.	B	Identification of inoperable CR annunciators

PWR RO Examination Outline

Facility: Byron | Exam Date: 9/14/98 | Examination Level: RO

Section Title Plant Systems

RO Group 1

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Control Rod Drive System	001 A2.06	3.4	Effects of transient xenon on reactivity	B	Effect of Xenon Transient & compensation
	001 K1.03	3.4	CRDM	B	Application of DC Hold
Reactor Coolant Pump System	003 A1.06	2.9	PZR spray flow	B	RCP and Pzr spray operations
	003 K2.01	3.1	RCPS	R	RCP Breaker & interlocks
Chemical and Volume Control System	004 A3.11	3.6	Charging/letdown	R	Charging & letdown flows (including seal injection)
	004 A4.07	3.9	Boration/dilution	B	Calculation of RCS Boron dilution
	004 K6.01	3.1	Spray/heater combination in PZR to assure uniform boron concentration	R	Boron mixing
Engineered Safety Features Actuation System	013 A3.01	3.7*	Input channels and logic	B	CNMT Spray/Phase B
	013 K4.13	3.7	MFW isolation/reset	R	FW isolation - P14
Nuclear Instrumentation System	015 A2.02	3.1	Faulty or erratic operation of detectors or compensating components	B	SR NIS discriminator failure
	015 K2.01	3.3	NIS channels, components, and interconnections	B	SR NIS - loss of control power
	015 K5.06	3.4	Subcritical multiplications and NIS indications	R	Eval for 1/M - Eightfold increase
In-Core Temperature Monitor System	017 K4.01	3.4	Input to subcooling monitors	R	CETC failure effect on Subcooling Monitor/Iconic Display
Containment Cooling System	2.1.32	3.4	Ability to explain and apply all system limits and precautions.	R	RCFC operations requirements
Main Feedwater System	2.1.7	3.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	B	SG Level program - low power
	059 K1.04	3.4	S/GS water level control system	R	Effect of failure of SG steam pressure channel
Auxiliary / Emergency Feedwater System	061 A3.01	4.1	AFW startup and flows	B	AFW Startup

PWR RO Examination Outline

Facility: Byron | Exam Date: 9/14/98 | Examination Level: RO

Section Title Plant Systems

RO Group 1

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Auxiliary / Emergency Feedwater System	061 K5.02	3.2	Decay heat sources and magnitude	B	AFW flow requirements for cooldown
Liquid Radwaste System	068 A4.04	3.8	Automatic isolation	B	RCDT operation - effect of CNMT Isolation
	068 K1.07	2.7	Sources of liquid wastes for LRS	R	CNMT Sump sources of input during normal operations
Waste Gas Disposal System	071 A4.05	2.6*	Gas decay tanks, including valves, indicators, and sample line	R	Waste Gas Decay Tank Operations
Area Radiation Monitoring System	072 A4.03	3.1	Check source for operability demonstration	R	Check Source operation
	072 K3.02	3.1	Fuel handling operations	B	Loss of FHB Overhead Crane rad monitor

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Plant Systems

RO Group 2

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Reactor Coolant System	002 A1.11	2.7	Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling	B	Relationship of levels during refueling operations
	002 A3.01	3.7	Reactor coolant leak detection system	R	RCS leak Detection Systems
	002 K4.09	3.2	Operation of loop isolation valves.	R	Use of Loop Isolation Valves
Emergency Core Cooling System	006 A2.13	3.9	Inadvertent SIS actuation	B	Systems response to SI/Actions
	006 K3.02	4.3	Fuel	B	10CFR50.46 Design Criteria
	006 K6.03	3.6	Safety Injection Pumps	B	Evaluation of flow ECCS pumps
Pressurizer Pressure Control System	010 A1.08	3.2	Spray nozzle DT	B	Spray using Normal and Aux Spray
	010 K5.01	3.5	Determination of condition of fluid in PZR, using steam tables	B	Evaluation of Pzr conditions
Pressurizer Level Control System	011 K1.04	3.8	RPS	B	Pzr Level Reactor Trip
Reactor Protection System	012 A3.07	4.0	Trip breakers	R	Operation of BOTH Bypass Trip Breakers
	012 A4.03	3.6	Channel blocks and bypasses	B	Input that can be bypassed & plant conditions
	012 K5.01	3.3*	DNB	R	OTdT inputs & effect of changes
Rod Position Indication System	2.4.31	3.3	Knowledge of annunciators alarms and indications, and use of the response instructions.	R	ROD BOTTOM Alarm operation
Non-Nuclear Instrumentation System	016 K3.02	3.4*	PZR LCS	B	NR RTD Failure effects
Containment Spray System	026 A2.08	3.2	Safe securing of containment spray when it can be done)	B	Sequence for securing CNMT Spray
	026 A4.01	4.5	CSS controls	R	Pump operation interlocks
Spent Fuel Pool Cooling System	033 K1.05	2.7*	RWST	R	RWST Purification Loops

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Plant Systems

RO Group 2

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
D.C. Electrical Distribution	2.1.30	3.9	Ability to locate and operate components, including local controls.	B	DC bus battery charger
Emergency Diesel Generators	064 A3.07	3.6*	Load sequencing	B	Sequencing of CNMT Spray pumps - SI & SI w LOP
Fire Protection System	086 K4.06	3.0	CO2	B	Effect of loss of DC - CO2 actuation

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Plant Systems

RO Group 3

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Residual Heat Removal System	005 K1.12	3.1	Safeguard pumps	B	Recirc interties to SI Pumps & CV Pumps
	005 K4.10	3.1	Control of RHR heat exchanger outlet flow	R	Failure of Hx Outlet Valve
Pressurizer Relief Tank/Quench Tank System	2.4.50	3.3	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	R	PRT conditions causing alarm/response
Component Cooling Water System	008 A2.05	3.3*	Effect of loss of instrument and control air on the position of the CCW valves that are air operated	R	Determination of effect of valve positioning
Containment Iodine Removal System	027 A4.03	3.3*	CIRS fans	R	Charcoal Filters response to deluge
Steam Dump System and Turbine Bypass Control	041 A3.02	3.3	RCS pressure, RCS temperature, and reactor power	B	Steam Dump input malfunction
Main Turbine Generator System	045 K1.20	3.4	Protection system	R	Turbine Control response to Failed Impulse Channel
Instrument Air System	078 K3.02	3.4	Systems having pneumatic valves and controls	B	Evaluation of eqpt affected for slow loss of IA

PWR RO Examination Outline

Facility: Byron   Exam Date: 9/14/98   Examination Level: RO

Section Title Emergency and Abnormal Plant Evolutions

RO Group 1

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Reactor Coolant Pump Malfunctions	015 AA2.10	3.7	When to secure RCPs on loss of cooling or seal injection	B	Eval loss of cooling flow (CCW)
	015 AK2.07	2.9	RCP seals	B	Eval of RCP seal failure
Emergency Boration	024 AA2.05	3.3	Amount of boron to add to achieve required SDM	B	Time/amount E-boration for condition
Loss of Component Cooling Water	026 AA1.05	3.1	The CCWS surge tank, including level control and level alarms, and radiation alarm	B	Evaluation of CCW leak
Pressurizer Pressure Control Malfunction	027 AA1.01	4.0	PZR heaters, sprays, and PORVs	B	Pressure controller step change
	027 AA2.15	3.7	Actions to be taken if PZR pressure instrument fails high	B	Non-Controlling channel failure
Steam Line Rupture	040 AA1.01	4.6	Manual and automatic ESFAS initiation	B	Steamline isolation
	040 AK1.06	3.7	High-energy steam line break considerations	B	Eval of Leak
Loss of Condenser Vacuum	051 AA2.02	3.9	Conditions requiring reactor and/or turbine trip	B	Eval of conditions
Station Blackout	055 EK3.02	4.3	Actions contained in EOP for loss of offsite and onsite power	B	Identification of RCP seal LOCA/cooldown
Loss of Vital AC Instrument Bus	057 AA2.19	4.0	The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus	B	Eqpt affected on bus loss
Control Room Evacuation	068 AA1.21	3.9	Transfer of controls from control room to shutdown panel or local control	B	Operations required for transfer
Inadequate Core Cooling	074 EK1.03	4.5	Processes for removing decay heat from the core	B	Major action categories
High Reactor Coolant Activity	076 AA2.02	2.8	Corrective actions required for high fission product activity in RCS	B	Actions for reducing activity
Pressurized Thermal Shock	E08 EK2.2	3.6	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	B	Identification of heat removal process

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Emergency and Abnormal Plant Evolutions

RO Group 1

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Natural Circulation Operations	E09 EK3.1	3.3	Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.	B	Natural Circ conditions and limits

PWR RO Examination Outline

Facility: Byron Exam Date: 9/14/98 Examination Level: RO

Section Title Emergency and Abnormal Plant Evolutions

RO Group 2

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Continuous Rod Withdrawal	001 AA2.05	4.4	Uncontrolled rod withdrawal, from available indications	B	Evaluate conditions - unwarranted rod withdrawal
Dropped Control Rod	003 AK3.10	3.2?	RIL and PDIL	B	P/A vs. Group Step Counters
Reactor Trip	007 EA1.03	4.2	RCS pressure and temperature	B	Stabilized RCS temperature with failure of Steam Dumps
	007 EK2.03	3.5	Reactor trip status panel	R	Reactor Trip requirements
Pressurizer Vapor Space Accident	008 AK1.01	3.2	Thermodynamics and flow characteristics of open or leaking valves	R	Tail-Pipe conditions
Small Break LOCA	009 EA1.10	3.8*	Safety parameter display system	B	Calculation of subcooled margin on Iconic Display
Large Break LOCA	011 EA1.03	4.0	Securing of RCPs	B	RCP trip criteria evaluation
Loss of Reactor Coolant Makeup	022 AA1.08	3.4	VCT level	B	VCT level transmitter malfunction
Loss of Residual Heat Removal System	025 AK1.01	3.9	Loss of RHRS during all modes of operation	B	Calc of time to saturation/core boiling
	025 AK3.01	3.1	Shift to alternate flowpath	B	Alternate RCS cooling
Anticipated Transient Without Scram	2.4.48	3.5	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.	B	AMS conditions
Loss of Source Range Nuclear Instrumentation	032 AK1.01	2.5	Effects of voltage changes on performance	B	Evaluation of SR NIS voltage failure
Loss of Intermediate Range Nuclear Instrumentation	033 AA2.04	3.2	Satisfactory overlap between source-range, intermediate-range and power-range instrumentation	B	Eval of failed IR channel on SU
Steam Generator Tube Leak	037 AA1.02	3.1*	Condensate exhaust system	R	Monitors for SG Tube leakage
Steam Generator Tube Rupture	038 EK3.06	4.2	Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures	B	Loss of subcooling

PWR RO Examination Outline

Facility: Byron　　Exam Date: 9/14/98　　Examination Level: RO

Section Title Emergency and Abnormal Plant Evolutions

RO Group 2

System/Evolution	K/A		RO	KA Statement	Level	Question Topic
Loss of Secondary Heat Sink	E05	EK2.1	3.7	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	B	Interlocks affecting reestablishment of feed
Loss of Emergency Coolant Recirculation	E11	EA1.1	3.9	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	B	Reason for rapid SG depressurization

PWR RO Examination Outline

Facility: Byron   Exam Date: 9/14/98   Examination Level: RO

Section Title Emergency and Abnormal Plant Evolutions

RO Group 3

System/Evolution	K/A	RO	KA Statement	Level	Question Topic
Pressurizer Level Control Malfunction	028 AK3.05	3.7	Actions contained in EOP for PZR level malfunction	B	Failed level channel low.
Loss of Off-Site Power	056 AA1.21	3.3*	Reset of the ESF load sequencers	B	Reset of sequencer
	056 AA2.46	4.2	That the ED/Gs have started automatically and that the bus tie breakers are closed	B	Eval of electric bus status