August 7, 2000

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON - NRC EXAMINATION REPORT 50-454/2000301(DRS); 50-455/2000301(DRS)

Dear Mr. Kingsley:

On June 29, 2000, the NRC completed initial operator licensing examinations at your Byron Nuclear Plant, Units 1 and 2. The enclosed report presents the results of the examination.

Four applicants were administered senior reactor operator license examinations and five applicants were administered reactor operator license examinations. The license applicants' performance evaluations were finalized on July 27, 2000. All applicants passed all sections of their examinations and were issued their respective operator licenses.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this examination.

Sincerely

/RA/

David E. Hills, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-454; 50-455
License Nos. NPF-37; NPF-66

Enclosures:
1. Operator Licensing Examination Report 50-454/2000301(DRS); 50-455/2000301(DRS)
2. Facility Comments and NRC Resolutions
3. Simulation Facility Report
4. Written Examination and Answer Keys (RO)
5. Written Examination and Answer Keys (SRO)

See Attached Distribution

cc w/encls 1, 2, 3: D. Helwig, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President, Nuclear Operations
H. Stanley, Vice President, Nuclear Operations
R. Krich, Vice President, Regulatory Services
DCD - Licensing
W. Levis, Site Vice President
R. Lopriore, Station Manager
B. Adams, Regulatory Assurance Manager
M. Aguilar, Assistant Attorney General
State Liaison Officer
State Liaison Officer, State of Wisconsin
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, 4, 5: D. Spoerry, Training Department

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DOCUMENT NAME: G:DRS\BYR2000301DRS.WPD

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State Liaison Officer
State Liaison Officer, State of Wisconsin
Chairman, Illinois Commerce Commission

cc w/encls 1, 2, 3, 4, 5: D. Spoerry, Training Department

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-454; 50-455
License Nos:	NPF-37; NPF-66
Report No:	50-454/2000301(DRS); 50-455/2000301(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Byron Nuclear Plant, Units 1 and 2
Location:	4450 North German Church Road Byron, IL 61010
Dates:	June 20–29, 2000
Inspectors:	A. M. Stone, Chief Examiner H. Peterson, Examiner S. Dennis, Examiner G. Wilson, Observer
Approved by:	David E. Hills, Chief, Operations Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Examination Report 50-454/2000301, 50-455/2000301, on 06/20-06/29/2000; Byron Nuclear Plant; Units 1 and 2. Other Activities.

The announced operator licensing initial examination was conducted by regional examiners in accordance with the guidance of NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 8. No issues were identified.

<u>Examination Summary:</u>

- Four senior reactor operator applicants and five reactor operator applicants were administered the written examination and operating tests. All nine applicants passed all portions of their respective examinations and were awarded senior reactor operator licenses or reactor operator licenses (Section 4OA5.1).

Report Details

4. **OTHER ACTIVITIES (OA)**

4OA5 Other

.1 Initial Licensing Examinations

a. Inspection Scope

The NRC examiners conducted announced operator licensing initial examinations during the week of June 20, 2000. The NRC developed the written examinations, job performance measures, and four dynamic scenarios. The facility licensee developed an additional three dynamic scenarios. Four senior reactor operator applicants and five reactor operator applicants received written examinations and operating tests.

b. Findings

Written Examination:

The written examination was administered on June 29, 2000, in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 8. The NRC examiners independently graded the written examination and concluded that all applicants achieved the passing criteria of 80 percent, with an average score of about 90 percent. The licensee submitted two post-examination comments on the written examination. The comments and the NRC's resolutions are contained in Enclosure 2 of this report. The licensee also identified the following generic performance deficiencies:

- Question #1 (RO) - four applicants were unable to specify entry conditions into BOA PRI-2, "Emergency Boration."
- Question #12 (RO) - three applicants were unable to identify the worst case accident for peak containment pressure for Unit 2.
- Question #31 (RO)/ #6 (SRO) - five applicants were unable to determine the cause of a steam line isolation given a set of initial conditions.
- Question #82 (RO)/ #57 (SRO) - five applicants were unable to determine the minimum boron concentration in the Spent Fuel Pool assuming all of the new Holtec storage racks were filled with irradiated fuel assemblies.
- Question #88 (SRO) - two applicants were unable to apply Technical Specification 3.8.1 to determine when a unit shutdown was required.
- Question #90 (SRO) - three applicants concluded that all steam generators were required to be at 10 percent narrow range level prior to manually isolating auxiliary feedwater to ruptured steam generator.

Operating Test:

The NRC determined that the three scenarios submitted by the facility were within the range of acceptability expected for the proposed examination. The NRC examiners did not identify any significant security concerns associated with the development or administration of the tests. During NRC validation of the tests, the examiners identified several procedural discrepancies. The facility licensee initiated a condition report to address these discrepancies.

The NRC examiners administered the operating tests during the week of June 20, 2000. All applicants demonstrated satisfactory performance during the dynamic scenarios and job performance measures and administrative tasks. Two generic performance deficiencies were identified with respect to taking manual action when automatic actions failed to occur. Several applicants identified that the automatic actions did not occur; however, did not take manual action until directed by the unit supervisor (senior reactor operator applicant). In some instances, manual actions were taken without prompting; however, the applicants did not announce that equipment failed to automatically actuate.

The NRC examiners also identified several individual deficiencies in applicant performance during the operating examination which are described in each individual's examination report, Form ES-303-1, "Operator Licensing Examination Report." The NRC forwarded copies of the evaluations under separate correspondence to the Site Training Manager.

In addition, a potential emergency procedure deficiency was identified. The NRC scenario #4 included a loss of coolant accident outside containment and was administered to two crews. The examiners noted that Step 30.c of 1BEP-0, "Reactor Trip or Safety Injection," required operators to determine whether reactor coolant system pressure was increasing or stable. If increasing, the operator was directed to proceed to Step 30.d then terminate safety injection. If reactor coolant pressure was decreasing, the operator was directed to continue with Step 31 and eventually to Step 37. This step required the operators to transfer to BCA1.2, "LOCA Outside Containment" if there was a loss of inventory outside containment and auxiliary building radiation levels were not normal. It was expected that the applicants would perform Step 37 of 1BEP-0 and isolate the leak. During both administrations, plant conditions were such that reactor pressure was increasing. Therefore, the applicants were not directed by the emergency procedures to address the leak outside of containment. The facility issued condition report B2000-018929 to determine: (1) if the current procedure is in compliance with the Westinghouse Owner's Group guidance; and (2) if a revision to the emergency procedures is necessary. This procedure discrepancy is considered an unresolved item (50-454/2000301-01; 50-455/2000301-01) pending review of the licensee's condition report.

4OA6 Meetings (Including Exit Meeting)

.1 Exit Meeting Summary

The inspectors presented the preliminary examination observations to Mr. Wozniak and other members of licensee management on June 29, 2000. The licensee acknowledged the issues presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

*B. Adams, Regulatory Assurance Manager
R. Deppi, Site Nuclear Oversight Manager
S. Gackstetter, Shift Operations Superintendent
J. Heaton, Initial License Training Specialist Lead
T. Horan, Operations Training Supervisor
*D. Spoerry, Training Manager
G. Stauffer, Regulatory Assurance
D. Wozniak, Engineering Director

NRC

B. Kemker, Resident Inspector

The above individuals also attended the exit interview conducted on June 29, 2000.

*Notified on August 1, 2000 of new unresolved item.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-454/2000301-01; 50-455/2000301-01	URI	Potential procedure discrepancy with respect to a loss of coolant accident outside of containment

Closed

None

Discussed

None

Facility Comments and NRC Resolutions

Written Examination Comments:

1. Question #33 (RO)/ #8 (SRO)

Byron is considered an Alternate AC (AAC) Station for design analysis during a Station Blackout. Which of the following is a reason that Byron chose to qualify as an AAC station instead of a 4-hour coping station?

A. EDG's have a high availability rate and are 100 percent redundant.

B. AAC source is available within 30 minutes.

C. AAC source is capable of carrying the immediate needs of both units for 8 hours.

D. Capable of unit cross-tying of auxiliary power from the Main Control Room.

Answer: D

Facility Comment:

Accept distractor A as an alternate answer. Distractor A is stated "EDG's have a high availability rate and are 100 percent redundant." One reason the site qualifies as an AAC station is that the EDGs have a reliability rate of .95 and are 100 percent redundant. The wording of this distractor is such that a high availability rate could be interpreted to be a realiability of >.95.

NRC Resolution:

Distractor A is not an acceptable answer. Distractor D is the only correct answer. The facility did not provide additional information regarding the terms "available" and "reliable." The applicants did not question the terminology during the examination. The licensee is correct in stating that a reliability rate of .95 was assumed in qualifying as an AAC station. Reliability is based on the probability that an EDG will start when given a start signal. Availability time is based on whether the EDG is capable of performing its safety function. For example, maintenance or surveillance activities may cause the EDG to be inoperable and reduce the availability time. The terms are not synonymous. It is possible to have a high availability rate but a low reliability rate - that is, an EDG fails to start on demand but the repair time is short.

Question History:

Distractor A originally included an incorrect reliability rate which required the applicant to memorize trivial data. The facility agreed to modify distractor A by replacing the specific values with "high availability."

Facility Comments and NRC Resolutions

2. Question #74 (RO)/ #29 (SRO)

Unit 1 was operating at 28 percent power when the Loop B Reactor Coolant Pump (RCP) tripped on overcurrent.

Which of the following describes the unit's initial response? (Assume NO operator action AND NO rod motion.)

A. A reactor trip occurs and unaffected loop Tave increases.

B. A reactor trip occurs and unaffected loop Tave decreases.

C. A reactor trip will NOT occur and unaffected loop Tave decreases.

D. A reactor trip will NOT occur and unaffected loop Tave increases.

Answer: C

Facility Comment:

Accept distractor D as the only correct answer. The question asks for the initial response of the unit due to a trip of a reactor coolant pump. Hot leg temperature in the unaffected loops will increase since the same reactor heat input is being absorbed by less total core flow. This caused Tave to initially increase.

NRC Resolution:

Change correct answer from distractor C to distractor D. The facility provided adequate documentation to support distractor D as the correct answer. The examiner agreed with the facility's reasoning and also noted that Figure HT-7-22C, "RCP B Trip, Manual Rods" (Heat Transfer Module), showed that Tave in the unaffected loops increase while Tave in the affected loop (Loop B) initially decreases.

History:

During their first review, the facility requested "unaffected loop" be changed to "Loop B" since rod motion could affect temperature in the unaffected loops. The NRC added "Assume NO operator action AND NO rod motion" to eliminate this concern and did not change the distractors.

SIMULATION FACILITY REPORT

Facility Licensee: Byron Nuclear Plant, Units 1 and 2

Facility Licensee Docket No: 50-454; 50-455

Operating Tests Administered: June 20–26, 2000

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION

1. None

Enclosure 4

WRITTEN EXAMINATION AND ANSWER KEY (RO)

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / $\boxed{\text{III}}$ / IV
Date: June 29, 2000	Facility/Unit:
License Level: $\boxed{\text{RO}}$ / SRO	Reactor Type: $\boxed{\text{W}}$ / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	__________ Points
Applicant's Score	__________ Points
Applicant's Grade	__________ Percent

Byron RO Written Examination
Answer Key

1	A	26	D	51	C	76	C
2	B	27	C	52	C	77	A
3	B	28	A	53	B	78	D
4	A	29	B	54	D	79	B
5	C	30	D	55	C	80	C
6	D	31	A	56	C	81	D
7	C	32	C	57	D	82	C
8	B	33	D	58	C	83	D
9	A	34	A	59	B	84	C
10	D	35	B	60	A	85	A
11	D	36	D	61	A	86	D
12	B	37	C	62	D	87	D
13	D	38	C	63	B	88	B
14	C	39	B	64	D	89	B
15	D	40	D	65	D	90	A
16	B	41	C	66	C	91	C
17	A	42	A	67	D	92	D
18	A	43	B	68	C	93	D
19	D	44	B	69	A	94	C
20	B	45	C	70	C	95	A
21	A	46	D	71	D	96	B
22	D	47	A	72	C	97	C
23	A	48	D	73	B	98	A
24	B	49	A	74	~~C~~ D	99	C
25	C	50	B	75	B	100	C

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
1	A	000024K3.01	4.1	2	F	New	BOA PRI-2
2	B	000076K201	2.6	2	F	Bank	System Description: RADIATION MONITORS
3	B	001K4.02	3.8	2	F	Bank	Rod Control Lesson Plan
4	A	003A2.01	3.5	2	F	Bank	RCP Lesson Plan
5	C	004K6.17	4.4	3	F	New	PRI-A Emergency Boration
6	D	004K3.07	3.8	3	H	New	PZR Lesson Plan
7	C	013A1.05	3.4	3	H	Bank	SSPS Lesson Plan
8	B	000067K1.01	2.9	2	F	New	Fire Protection Lesson Plan
9	A	015A1.01	3.5	3	H	New	Nuclear Instrumentation Lesson Plan
10	D	015A3.04	3.3	2	F	Bank	Nuclear Instrumentation Lesson Plan
11	D	061A1.05	3.6	3	H	Bank	BOP AF-7 AFW Lesson Plan
12	B	000069A1.02	3.3	2	F	New	EF-4 ESF Lesson Plan
13	D	000009K3.24	4.1	2	H	New	1 BEP-1
14	C	000032A1.01	3.1	4	H	Bank	1 BGP 100-5
15	D	W/E05A2.01	3.4	2	F	Bank	1 BFR-H1
16	B	012A2.04	3.1	3	H	Bank	SSPS Lesson Plan
17	A	039A2.04	3.4	3	H	Bank	Steam Dump Lesson Plan
18	A	063K3.02	3.5	3	H	New	DC-1 DC Power
19	D	G2.2.13	3.6	2	F	Bank	BAP 330-1
20	B	005A2.02	3.5	3	H	Bank	PZR Lesson Plan BCB-1
21	A	000065A1.03	2.9	3	F	Bank	1BOA SEC-4
22	D	103000G2.1.12	2.9	2	F	Bank	SDM#40 “Containment” TS 3.6.1.4
23	A	01500K5.16	2.9	2	H	New	Nuclear Instrumentation Lesson Plan

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
24	B	075K1.02	2.9	2	F	Bank	Liquid Rad Waste Lesson Plan
25	C	002000K4.10	4.2	2	F	New	1BGP 100-5
26	D	000005K3.01	4.0	3	H	New	ES-0.1 Step 5
27	C	000015A1.22	4.0	4	H	Bank	1 BOA RCP-1
28	A	W/E09K2.02	3.6	3	H	Bank	LOCA Procedure Lesson Plan
29	B	000026A2.01	2.9	4	H	Bank	BOA PRI-6 Att A CC Lesson Plan
30	D	000027A1.01	4.0	3	H	New	BAR 1-12-A1
31	A	000040K3.02	4.4	3	H	Bank	SSPS Lesson Plan
32	C	000051A2.02	3.9	3	H	New	1 BOA SEC-3
33	D	000055G2.1.10	2.7	2	F	New	ECA 0.0 Lesson Plan
34	A	000057G2.4.10	3.0	3	H	Bank	1BOP ELEC-2
35	B	000062A1.01	3.1	2	H	New	Containment Ventilation Lesson Plan ESW Lesson Plan 1BOA PRI-7
36	D	013A4.02	4.3	3	F	Bank	FW-1 FW Lesson Plan
37	C	000068A2.09	4.1	3	H	New	Steam Tables
38	C	061K1.02	3.4	3	F	New	AFW Lesson Plan
39	B	000074K1.03	4.5	3	F	Bank	1 BFR-C.1 Procedure Lesson Plan
40	D	G2.1.1	3.7	2	F	New	BAP 300-1 Conduct of Ops
41	C	G2.1.3	3.0	2	F	Bank	OP-AA-101-401
42	A	G2.1.12	2.9	3	H	New	ITS 3.4.13
43	B	001K5.05	2.8	4	H	New	1BCB-1 Tables
44	B	003A1.09	2.8	3	H	Bank	RCP Lesson Plan
45	C	004A4.18	4.3	3	F	New	CVCS Lesson Plan ITS Boration flow path
46	D	013A3.02	4.1	3	H	Bank	EF-2 ESF setpoint
47	A	000038K3.08	4.2	2	F	Bank	1BEP-3

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
48	D	017K6.01	2.7	2	F	Bank	Incore Instrumentation Lesson Plan
49	A	022G2.1.32	3.4	2	F	Bank	ITS 3.6.5
50	B	056A2.04	2.6	4	H	New	MFW Lesson Plan
51	C	059K4.19	3.2	3	H	Bank	MFW Lesson Plan
52	C	061K2.03	4.0	2	F	New	AF-1 AFW Lesson Plan
53	B	068K4.01	3.4	2	F	New	Liquid Rad Waste Lesson Plan
54	D	071K4.01	2.6	2	F	New	RW-1 Gaseous Radwaste Lesson Plan
55	C	072G2.1.14	2.5	2	H	Bank	Control Room HVAC Lesson Plan ITS 3.3.6
56	C	G2.2.11	2.5	2	F	New	CC-AA-112, Temp Mods
57	D	G2.2.12	3.0	2	F	Bank	ITS SR 3.0.2
58	C	000001A2.05	4.4	3	H	Bank	1BOA ROD-1
59	B	000003G2.2.4	2.8	2	H	New	1BOA ROD-3
60	A	064K4.11	3.5	3	H	Bank	AFW Lesson Plan EF-1 ESF setpoints
61	A	000008A2.12	3.4	3	F	New	PZR Lesson Plan
62	D	W/E04K1.02	3.5	2	F	New	1 BCA-1.1
63	B	W/E03K3.03	3.9	2	F	New	1BEP ES-1.2
64	D	W/E11K2.01	3.6	2	F	New	ECCS Lesson Plan ECCS-3
65	D	000022A2.02	3.2	3	H	Bank	1 BOA PRI-1
66	C	000025K3.01	3.1	3	H	New	1 BOA PRI-10
67	D	000029A1.15	4.1	3	H	New	AMS Lesson Plan
68	C	000037A2.01	3.0	4	H	Bank	1 BOA SEC-8
69	A	000054K1.01	4.1	3	H	New	1BEP-2 Background info EP-2
70	C	000007K2.02	2.6	2	H	New	BAR 1-11-B5
71	D	078K3.02	3.5	3	H	Bank	Instrument Air Lesson Plan

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
72	C	G2.3.1	2.6	3	H	Bank	NGET
73	B	G2.3.4	2.5	3	H	Bank	NGET
74	~~C~~ D	002K6.02	3.6	2	F	New	ESF Setpoint EF-1 Figure HT-7-22C
75	B	006K2.04	3.6	3	H	New	ECCS Lesson Plan
76	C	010A3.02	3.6	3	H	New	PZR Lesson Plan
77	A	011K5.12	3.2	2	F	Bank	PZR Lesson Plan
78	D	014K5.01	2.7	4	H	New	Rod Position Lesson Plan TS bases
79	B	016K3.02	3.4	3	H	New	BFR-I.3 Lesson Plan BEP ES-0.2 BEP ES 1.2
80	C	026A4.01	4.5	3	H	Bank	CS Lesson Plan
81	D	029G2.1.27	2.8	2	F	New	VP-2 Containment Purge
82	C	033K4.05	3.1	2	F	New	TS 3.7.15
83	D	035K5.03	3.6	3	F	Bank	SG Lesson Plan
84	C	055K3.01	2.5	2	F	New	SEC-3 Condenser Vacuum
85	A	062K2.01	3.3	2	F	New	AC Power Lesson Plan
86	D	086A2.02	3.0	2	F	Bank	Fire Protection Lesson Plan
87	D	073K1.01	3.6	3	H	New	Component Cooling Lesson Plan
88	B	G2.3.10	2.9	3	H	New	NGET
89	B	G2.4.1	4.3	2	H	New	1BFR-H.1
90	A	00028K2.03	2.6	3	H	New	PZR Lesson Plan
91	C	000056K1.01	3.7	2	F	Bank	1BCA-0.1
92	D	000011G2.4.18	2.7	2	F	New	1 BEP-1 fold out page
93	D	G2.4.2	3.9	2	F	Bank	1BEP ES1.3 ESF Lesson Plan
94	C	G2.4.3	3.5	4	F	New	ITS 3.3.i
95	A	007A2.01	3.9	2	F	New	1BEP-0

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
96	B	008K3.01	3.4	4	H	New	1BOA PRI-6 Att A
97	C	028A4.01	4.0	3	F	New	0 BOSR 6.8.1-1
98	A	079G2.1.28	3.2	2	F	New	Service Air Lesson Plan
99	C	041K4.09	3.0	3	H	Bank	Steam Dump Lesson Plan
100	C	G2.4.6	3.1	2	F	New	FR-C1 Procedure Lesson Plan

Question #1

Which of the following conditions does NOT require entry into BOA PRI-2 “Emergency Boration” ?

A. Keff > 0.95 during Mode 5

B. Inadequate shutdown margin

C. Uncontrolled cooldown with the reactor shutdown

D. 3 RCCA did not fully insert following a reactor trip

Question #2

The failed fuel monitor 1RT-PR006 uses which of the following types of detectors?

A. Geiger-Mueller (G-M) tube.

B. NaI crystal scintillation.

C. Compensated Ion Chamber.

D. BF_3 Neutron detector.

Question #3

Given the following plant conditions:

Reactor power is 75%
Control rods can not be moved in AUTO or MANUAL due to a failure.

Which function is impaired if control bank D rods were moved using BANK SELECT?

A. The pulse to analog converter display for bank D.

B. Bank overlap function when control rods are inserted.

C. Rod insertion limit alarms when inserting control rods.

D. Control rod stop alarm actuation when reaching C-11.

Question #4

Which of the following is the reason for promptly closing the seal leakoff isolation valve for a RCP with a HIGH number 1 seal leakoff once the RCP has stopped rotating?

A. Protect number 2 seal from possible debris from the number 1 seal.

B. Prevention of damage to the thermal barrier due to high flow.

C. Minimize the amount of RCS water that is routed to containment sump.

D. Assure a minimum back pressure is maintained on the number 3 seal.

Question #5

While performing an emergency boration in accordance with PRI-2, which of the following is the correct order for boration methods?

A. (1) Emergency Borate Valve 1CV8104 from MCR OR RWST valves 1CV112D, 1CV112E
(2) Normal Borate Valves 1CV110A, 1CV110B
(3) Manually operate Emergency Borate Valve 1CV8104 from 426 VCT valve aisle

B. (1) RWST valves 1CV112D, 1CV112E OR Emergency Borate Valve 1CV8104 from MCR
(2) Normal Borate Valves 1CV110A, 1CV110B
(3) Manually operate Emergency Borate Valve 1CV8104 from 426 VCT valve aisle

C. (1) Normal Borate Valves 1CV110A, 1CV110B OR Emergency Borate Valve 1CV8104 from MCR
(2) RWST valves 1CV112D, 1CV112E
(3) Manually operate Emergency Borate Valve 1CV8104 from 426 VCT valve aisle

D. (1) Manually operate Emergency Borate Valve 1CV8104 from 426 VCT valve aisle OR Normal Borate Valves 1CV110A, 1CV110B
(2) RWST valves 1CV112D, 1CV112E
(3) Emergency Borate Valve 1CV8104 from MCR

Question #6

The unit was initially at 100% equilibrium power with all systems in automatic. A plant transient caused Pressurizer level to increase to 68% and pressure to increase to 2280 psig.

Which of the following describes the Pressurizer heaters and spray status for these conditions?

	Backup Heaters	Variable Heaters	Spray Valves
A.	ON	ON	CLOSED
B.	OFF	OFF	CLOSED
C.	OFF	ON	THROTTLED OPEN
D.	ON	OFF	THROTTLED OPEN

Question #7

RCS pressure has decreased to 1850 psig during a plant cooldown. The P-11 bypass permissive is LIT and appropriate actions have been taken as required by 1BGP100-4, "Plant Shutdown." Subsequently, a steamline break occurs **downstream** of the MSIV's.

What is the ESF response to this leak?

A. Dependent upon break size, both a steamline isolation and an SI will occur.

B. A steamline isolation will always occur but an SI will only occur on a large break.

C. Dependent upon break size, a steam line isolation will occur; however an SI will not occur.

D. An SI will always occur, but a steamline isolation will only occur on a large break.

Question #8

Which of the following is the correct classification of a fire in the Diesel Generator Fuel Day Tank?

A. Class A

B. Class B

C. Class C

D. Class D

Question #9

During the performance of a Calorimetric at 100% power, an operator uses a Feedwater Temperature 30 degrees LOWER than actual.

(1) Would the calculated value of power be HIGHER or LOWER than actual power?

(2) Based on the calculated power would an adjustment of the NIS Power Range Channels be CONSERVATIVE or NON-CONSERVATIVE with respect to protection setpoints?

	(1)	(2)
A.	higher	conservative
B.	higher	non-conservative
C.	lower	conservative
D.	lower	non-conservative

Question #10

How are the input signals used by the power range nuclear instrumentation Channel Comparator ?

A. Compares normalized signal of detector B (lower) to detector A (upper) and generates alarm when greater than a 4% difference.

B. Compares each lower detector to the average of the lower detectors and each upper detector to the average of the upper detectors and generates an alarm when greater than 4% difference.

C. Compares total power from each channel to average power and generates an alarm when any one channel is greater than 3% of average.

D. Compares total power from each channel to lowest total power value from all channels and generates an alarm at 3% difference.

Question #11

Unit 1 in MODE 3, "Diesel Driven AFW Pump Monthly Surveillance", is in progress.

The following conditions are noted with respect to the 1B AFW pump:

Suction pressure..........17 psig
Discharge pressure.......1900 psig
Engine Speed................1910 rpm
Recirc Flowrate.............90 gpm
ALL SG levels slowly INCREASING.

Which of the following describes the operator actions required by these conditions ?

A. Dispatch an operator to identify any leaks in the discharge header piping.

B. Verify the SX suction valves 1AF006B and 1AF017B are OPEN.

C. Dispatch an operator to check the position of recirc valves and locally verify recirc flow.

D. Trip the 1B Diesel Driven AFW pump.

Question #12

For **Unit 2**, the worst case accident for peak containment pressure would be a double ended guillotine break of the ____(1)____. The resultant peak containment pressure would be at __(2)__ psig.

Which of the following accidents and pressure are correct?

	(1)	(2)
A.	Pressurizer relief line	41.6 psig.
B.	RCS at the RCP suction	44.4 psig.
C.	Main steam line in containment	45.6 psig.
D.	Feedwater line in containment	47.8 psig.

Question #13

The following plant conditions exist for Unit 1:

Reactor Trip and Safety Injection have occurred following a LOCA
MSIVs have just closed due to Containment pressure
RCS subcooling is acceptable per ICONIC display

Which of the following conditions will allow ECCS to be reduced?

1. S/G narrow range levels 12% for all 4 S/G
2. S/G narrow range levels 32% for all 4 S/G

3. RCS pressure is decreasing
4. RCS pressure is stable

5. PZR level is 10%
6. PZR level is 40%

A. 1,3,5

B. 1,4,5

C. 2,3,6

D. 2,4,6

Question #14

A reactor startup was aborted at 10E-8 amps due to severe weather conditions.

Plant conditions are as follows:

All control banks have been inserted
The reactor trip breakers are closed
Intermediate range channels N35 and N36 read 1E-11 amps
Source range channels N31 and N32 are deenergized

Which of the following operator actions are required to energize BOTH source range channels?

A. De-energize TWO power range channels by pulling the instrument power fuses on two of the power range channel drawers.

B. Place BOTH source range manual block switches to BLOCK.

C. Place BOTH source range manual block switches to RESET.

D. Place BOTH source range “High Flux at Shutdown” switches to the BLOCK position.

Question #15

Given the following plant conditions:

The plant has experienced an unisolable main steam line break inside containment. The operators are implementing actions of 1BCA-2.1 “Uncontrolled Depressurization of all S/G’s”. Feed flow was reduced to 25 gpm to each S/G by operator action.

Based on the above conditions, which of the following describes the use of 1BFR-H.1, “Loss of Secondary Heat Sink” .

The complete performance of 1BFR-H.1 is ___?

A. required immediately.

B. required when 10% NR level cannot be restored to ONE steam generator.

C. required when 10% NR level cannot be restored to ALL steam generators.

D. not required.

Question #16

Given the following plant conditions:

Reactor Power is 100%
Reactor trip breaker testing is being performed with Reactor Trip Bypass breaker A (BYA) racked in and closed
Both Reactor Trip Breakers (RTA and RTB) are closed

What would be the result if a failure of a single 15 VDC power supply in the “A” Train SSPS Logic cabinet occurred?

A. The redundant power supply maintains normal conditions and a Rod Dev Power Rng Tilt alarm is generated.

B. Plant conditions remain stable with the General Warning alarm remaining lit.

C. The reactor trips when both the UV and Shunt trip coils are actuated for RTA.

D. The reactor trips when the UV trip coils are actuated for both RTA and RTB.

Question #17

Given the following plant conditions:

A turbine runback was initiated from 100% power
Tave is 577°F and decreasing
Tref is 571°F and stable
GROUP I and II Steam Dump Valves are FULL OPEN
GROUP III and IV Steam Dump Valves are CLOSED
“Steam Dump Actuated” status light is NOT lit

Which of the following explains the status of the Steam Dump system?

Steam Dumps are operating ______

A. INCORRECTLY because the GROUP II Steam Dumps should be throttled open.

B. INCORRECTLY because the GROUP III Steam Dumps should be throttled open.

C. CORRECTLY because the HI-1 bistable remains locked in until C-7 is reset.

D. CORRECTLY because the operator is required to reset C-7 when Tave stabilizes.

Question #18

The plant was at 50% power with a normal electrical lineup. A loss of DC Bus 111 occurs. Assuming no operator action, which of the following will occur?

A. Reactor trip from low-low SG level.

B. Loss of field flashing for 1B diesel generator.

C. Turbine trip due to loss of power to the 20-2/AST solenoid.

D. Loss of Power to Bus 159 following Main Generator Trip.

Question #19

Which of the following would be an EXCEPTIONAL Out-Of-Service if single valve isolation is used?

The system has a temperature of _(1)_ OR a pressure of _(2)_.

	(1)	(2)
A.	150 °F	250 psig.
B.	170 °F	350 psig.
C.	190 °F	450 psig.
D.	210 °F	550 psig.

Question #20

Given the following conditions on Unit 1:

Unit is in MODE 5 during cooldown per BGP 100-5
RCS has just been filled to solid plant condition
RH pump 1A is operating in Shutdown Cooling mode
RCS temperature is 150°F and stable
RCS pressure is 335 psig and stable

A failure of the letdown pressure control valve controller, PK-131, causes RCS pressure to rise to 454 psig, with RH pump 1A delta-P measured at 120 psid.

Which of the following describes ALL the component actions that occur to mitigate the consequences of this pressure rise, assuming no operator action? (References are provided)

A. Both PZR PORV's will open; the RHR loop suction relief valve and the RHR loop 1A discharge relief will open.

B. PZR PORV 1RY455A and the RH loop suction relief valve will open.

C. RH loop suction relief valve and RH discharge relief valve will open.

D. PZR PORV 1RY456 will open and the RH loop suction relief valves from the RCS 1RH8701 and 1RH8702 will close.

Question #21

Unit 1 was at 400°F when the instrument air header depressurized. Immediate actions were taken in accordance with 1BOA SEC-4, "Loss of Instrument Air". The cause of the instrument air problem was quickly repaired and the standby service air compressor restored header pressure to above 95 psig. During restoration from this loss of instrument air, operators took manual actions to prevent unintended operation of equipment. Which of the following controllers would the operators place in MANUAL with a 0% demand to prevent unintended operation?

A. PZR Spray Valve Controllers, 1RY 455B and 1RY 455C

B. Charging Flow Controller, 1CV121

C. RH Heat Exchanger Bypass Flow Control Valves, 1RH618 and 1RH619

D. 1A/B Letdown Heat Exchanger Outlet Temperature Controller, 1CC130A/B

Question #22

The following plant condition exists:

Unit 1 is in HOT SHUTDOWN.

Which of the following is the MAXIMUM allowable Unit 1 containment internal pressure reading in accordance with Technical Specifications 3.6.4, "Containment Pressure"?

A. -.10 psig

B. -.50 psig

C. +.50 psig

D. +1.0 psig

Question #23

The following microampere readings were taken from the Power Range NIS detectors:

	N41	N42	N43	N44
Detector A (Upper)	360	330	345	325
Detector B (Lower)	360	365	360	375

The 100% current value for all detectors is 400 microamperes. Which of the following channels has the detector with the most limiting Quadrant Power Tilt Ratios?

A. N41

B. N42

C. N43

D. N44

Question #24

Which of the following conditions will cause radwaste key locked release tank outlet valves 0WX353 and 0WX896 to auto-close?

A. High alarm on 0PR10J, Station Blowdown Rad. Monitor.

B. Circulating water blowdown flow of 9500 gpm.

C. Both Inlet and Outlet valves of a release tank inadvertently opened.

D. Conductivity level of 0.22 micro-mhos on the outlet of the radwaste mixed bed demineralizer.

Question #25

The plant is in Mode 3. The crew is performing a plant cooldown in accordance with 1 BGP 100-5. Per Technical Specifications, the PZR PORV's must be placed in the ARM LOW TEMP position prior to decreasing RCS temperature below _(1)_ °F as long as the highest reading RCS pressure indicator (PT-403, 403A, and 405) is below the PORV Cold

Overpressure setpoint as determined by the __(2)__ temperature.

	(1)	(2)
A.	370	average RCS Tcold
B.	360	highest RCS Tcold
C.	350	lowest RCS Tcold
D.	340	average CETC

Question #26

The transition is made from EP-0 to ES-0.1 on Unit 1. Step 4 in ES-0.1 requires boration for all rods NOT fully inserted. There are 3 rods not fully inserted into the core at this point. What is the MINIMUM gallons that will have to be borated from the RWST for the three rods?

A. 1320 gallons

B. 5500 gallons

C. 3960 gallons

D. 16500 gallons

Question #27

Given the following conditions:

Unit 1 is operating at 100% power
RCP No. 1 SEAL LEAKOFF FLOW HIGH alarm is received
No. 2 seal leakoff high flow alarm has been printed
RCP No. 1 seal leakoff recorder indication is high offscale on the high range
Make-up to the RCS has increased 40 gpm to maintain PZR level

Which of the following has occurred and what action is required?

A. The No. 1 and No. 2 seals have failed and a controlled reactor shutdown is required.

B. Only the No. 2 seal has failed and continued monitoring of RCP conditions is required.

C. The No. 1 seal has failed and an immediate reactor trip is required.

D. The No. 2 and No. 3 seals have failed and continued monitoring of RCP conditions is required.

Question #28

During a small break LOCA on a cold leg, a phase is reached where the vessel level continues to decrease below the hot leg penetrations and boiling in the core is the means of transporting the core heat to the bubble. A fixed differential pressure exists between the core and the break and is maintained by the loop seal.

What is the primary mechanism for heat removal during this phase?

A. Condensation of vapor from the bubble at the hot leg side of the SG U-tubes which then drains back to the core via the hot legs.

B. Condensation of vapor in the head, which is cooled by fans in containment, and draining back to the core.

C. Slug flow via the cold legs through the loop seal and flashing across the cold leg break.

D. Partial natural circulation flow characterized by liquid pulses flowing from the cold leg over the U-tubes and into the hot legs.

Question #29

The following plant conditions exist:

The reactor is shutdown
RCS temperature is 290°F and stable
RCS pressure is 320 psig and stable
RH is in shutdown cooling
RH Letdown is in service
CC surge tank level is slowly decreasing with the makeup valves to CC surge tank fully open

A leak has occurred in the ?

A. RH Heat Exchanger

B. Seal Water Heat Exchanger

C. Letdown Heat Exchanger

D. Thermal Barrier Heat Exchanger

Question #30

Given the following conditions on Unit 1:

Reactor power is steady at 100%
Tave is steady at 582°F
PZR level is 60% and slightly increasing
PZR pressure is 2230 psig and slowly decreasing
ALL systems are aligned normally

Which of the following conditions has occurred?

A. LK-459 PZR level controller has failed high

B. PZR PORV 456 is full open

C. PZR pressure transmitter PT-458 has failed high

D. PZR spray valve RY455B has failed to 50% open

Question #31

Unit 1 has tripped due to a steamline break inside containment. Shortly after the trip, the following parameters were recorded:

PZR pressure 1750 psig and stable
PZR level 22% and stable
CNMT pressure 7.8 psig (on all instruments)

S/G level(NR)	31% A,	30% B	25% C	34% D
S/G pressures	760 psig 1A	775 psig 1B	680 psig 1C	800 psig 1D

A steamline isolation occurred due to ....?

A. S/G low pressure steamline isolation.

B. S/G pressure rate steamline isolation.

C. the containment pressure circuit for steamline.

D. the PZR low pressure SI.

Question #32

The plant has the following conditions:

Reactor Power 52% steady state
Generator load is steady at 600MW
Condenser vacuum 2.2 in.HgA and steady

A leak developed in one of the water boxes causing pressure to rise at the rate of 0.2 inches HgA/minute. After 2 minutes, the operator began a load decrease at the rate of 10MW/minute in an attempt to offset the pressure rise and reduce load below the P-8 setpoint.

Assuming the load decrease remained constant and the rate of pressure rise remained constant throughout the event, what action is required? (References are provided)

A. The operator would initiate a manual turbine trip after the load is reduced to less than 30%.

B. No operator action, the turbine will automatically trip at 35% power causing a reactor trip.

C. The operator will initiate a manual reactor trip at approximately 39% power.

D. The operator will initiate a manual reactor trip at approximately 47% power.

Question #33

Byron is considered an Alternate AC (AAC) Station for design analysis during a Station Blackout. Which of the following is a reason that Byron chose to qualify as an AAC station instead of a 4-hour coping station?

A. EDG's have a high availability rate and are 100% redundant

B. AAC source is available within 30 minutes.

C. AAC source is capable of carrying the immediate needs of both units for 8 hours

D. Capable of unit cross-tying of auxiliary power from the Main Control Room.

Question #34

An operator noted the following annunciators were in following an event: (Not all alarms are provided)

PWR RNG HIGH STPT RX TRIP ALERT
OPDT HIGH ROD STOP C-4
OTDT HIGH ROD STOP C-3
PZR PRESS CONT DEV LOW HTRS ON
RCP BUS UNDERVOLT RX TRIP ALERT
RCP 1C BRKR OPEN OR FLOW LOW ALERT
TURB STOP VLV CLOSED ALERT

Which bus lost power?

A. Instrument Bus 113

B. Instrument Bus 112

C. DC Bus 113

D. DC Bus 112

Question #35

Which of the following describes the effect on containment if the Essential Service Water supply to ALL Reactor Containment Fan Coolers (RCFC) is secured? (Assume normal 100% power operation.)

Containment temperature would...

A. remain the same since the other containment HVAC equipment would maintain cooling.

B. increase since Containment Chiller will also trip upon Essential Service Water isolation.

C. increase because only Essential Service Water supplies RCFC's.

D. increase since Component Cooling can only supply RCFC's with a manual lineup.

Question #36

The plant was operating at 50% power when an inadvertent safety injection occurred. The operators wish to regain control of feedwater valves in order to feed the steam generators using the startup feedwater pump.

Which of the following is the correct order of actions that will be successful in restoring control of feedwater valves:

A. Reset SI, reset FW Isolation, cycle reactor trip breakers, reset FW Isolation Aux relays

B. Reset FW Isolation, reset SI, cycle reactor trip breakers, reset FW Isolation Aux relays

C. Reset SI, cycle reactor trip breakers, reset FW Isolation Aux relays, reset FW Isolation.

D. Reset SI, cycle reactor trip breakers, reset FW Isolation, reset FW Isolation Aux relays.

Question #37

The Control Room has been evacuated in accordance with BOA PRI-5 and the operators are performing an RCS cooldown.

The plant conditions are as follows:

Reactor coolant temperature is 456°F and stable
Reactor coolant pressure is 449 psig and stable

Which of the following describes the approximate state of the Reactor coolant when checking subcooling margin? (References are provided)

It is about ....?

A. 3 degrees superheated

B. at the saturation point

C. 3 degrees subcooled

D. 12 degrees subcooled

Question #38

Which of the following describes the relationship between the Unit 1 Auxiliary Feedwater System (AFW) piping and the Main Feedwater System piping?

The AFW piping connects downstream of the...

A. MFW bypass valves 6-inch piping and upstream of the FWIV FW035A.

B. MFW regulating valves 14-inch piping and upstream of the FWIV FW034A.

C. FWIV FW035A and upstream of the containment penetration.

D. Containment penetration and upstream of the last feedwater check valve after FWIV FW034A and prior to the SG.

Question #39

Which of the following sets of actions states the proper sequence of major actions to be performed in accordance with 1BFR-C.1, “Response to Inadequate Core Cooling”, for removing heat from the core?

A. Restoration of ECCS flow
RCP restart
Rapid secondary depressurization

B. Restoration of ECCS flow
Rapid secondary depressurization
RCP restart

C. RCP restart
Restoration of ECCS flow
Rapid secondary depressurization

D. RCP restart
Rapid secondary depressurization
Restoration of ECCS flow

Question #40

A non-licensed individual may move control rods using the IN/HOLD/OUT switch located in the control room under which of the following conditions?

The non-licensed individual is ...

A. a plant operator performing a surveillance test and is directly supervised by the on shift NSO.

B. a qualified nuclear engineer performing a control rod test and is directly supervised by a previously licensed NSO for that unit.

C. a plant operator who is enrolled in the initial license training program and is directly supervised by a certified instructor of the class.

D. a maintenance manager who is enrolled in the initial license training program and is under the direct supervision of the on shift NSO.

Question #41

An NRC-licensed operator works shift Monday morning as an NSO for 8 hours on Unit 1. The same individual is off work on Tuesday. On Wednesday morning the same operator stands the Unit 1 NSO watch for 8 hours. The same individual is off of work on Thursday. On Friday night the same operator is assuming the Unit 1 NSO watch at shift turnover

What is the administrative procedural requirement associated with reviewing the Unit logs?

A. Thursday only.

B. Thursday and Friday only.

C. Wednesday, Thursday, and Friday only.

D. A minimum of the past five days.

Question #42

Given the following conditions on Unit 2:

Reactor Power is 100%

A leak rate surveillance indicates the following:
Total RCS leakage rate is 9.0 gpm
Leakage to PRT is 6.0 gpm
Leakage to Reactor Coolant Drain Tank is 2.0 gpm
Leakage into Secondary from Primary as follows:

Unit 2 A S/G .07 GPM
B S/G .08 GPM
C S/G .09 GPM
D S/G .10 GPM

Which of the following statements are correct concerning the above conditions?

A. No leakage limits have been exceeded.

B. Unidentified leakage limit has been exceeded.

C. Total Primary to Secondary leakage limit has been exceeded.

D. Secondary leakage limit through one S/G has been exceeded.

Question #43

Which of the following operations results in the largest reactivity change? (References are provided)

A. Inserting 10 steps with rods initially at 190 steps on CBD at 100% power at 50 EFPH.

B. Inserting 10 steps with rods initially at 190 steps on CBD at 0% power at 11,500 EFPH

C. Withdrawing 10 steps with rods initially at 190 steps on CBD at 100% power at 11,500 EFPH.

D. Withdrawing 10 steps with rods initially at 190 steps on CBD at 0% power at 50 EFPH

Question #44

How would the RCP seals be affected if 1CV8142, #1 Seal Bypass Valve, was opened with the associated RCP running at normal operating pressure in RCS?

A. Flow across the #1 seal will fall to 0 gpm and the seal will be damaged by overheating.

B. Differential pressure changes across the #1 seal resulting in unbalanced seal motion.

C. Full RCS pressure is applied to the #3 Seal causing it to become the primary seal.

D. Pressure to the seal return line to the VCT is lowered causing flow across #2 seal to drop.

Question #45

Why is the manual emergency boration valve, CV8439, not used for performing emergency boration?

A. There is no way to monitor flow through the valve when in use so total boration flow could not be determined.

B. The throttling characteristics of the valve are poor, thereby resulting in full flow of 75 gpm or no flow at all.

C. The valve will only allow 10 gpm flow thereby not meeting the criteria for emergency boration.

D. Locally operated valves are not analyzed for safety functions and thereby not considered for performing safety function.

Question #46

Given the following plant conditions on Unit 1:

Reactor power was at 100% when a spurious SI signal was generated
Reactor Trip Breaker B failed to open
The spurious SI signal was cleared
The RH pumps, SI pumps, and 1A CV pump were secured.

After the ECCS pumps were secured, a small break LOCA occurred.

Which of the following occurs when containment pressure rises to 10 psig? (Assuming no operator actions are taken)

A. Only the MSIV and MSIV bypass valves close.

B. 1B and 1C MSIV's close but the 1A and 1D MSIV's remain open.

C. The 1A RH, 1A SI, and 1A CV Pumps start; the MSIV and MSIV bypass valves close.

D. The 1B RH and 1B SI Pumps start; the MSIV and MSIV bypass valves close.

Question #47

Which of the following determines the target temperature at which RCS cooldown is terminated following a S/G tube rupture using 1BEP-3, "Steam Generator Tube Rupture"?

A. The ruptured S/G pressure.

B. RCS subcooling of 39 degrees.

C. The lowest intact S/G pressure.

D. Maximum temperature for placing RH in service in the event of a loss of High Head Flow.

Question #48

A LOCA has occurred. Core exit thermocouple temperatures are indicating 690°F and increasing rapidly.

The Incore Thermocouples are providing satisfactory indication and will become __(1)__ accurate above __(2)__. (Assume NO core cooling is present)

	(1)	(2)
A.	less	700 °F
B.	more	1800 °F
C.	more	700 °F
D.	less	1800 °F

Question #49

The containment average temperature is the calculated average of the RCFC Dry Bulb _____.

A. inlet temperature of those RCFC's that are running.

B. outlet temperature of all RCFC's regardless of operating status.

C. inlet temperature of all RCFC's regardless of operating status.

D. outlet temperature of those RCFC's that are running.

Question #50

Given the following plant conditions on Unit 1:

Reactor power is 100%
3 CD/CB pumps are running
CD/CB Pump Selector Position is selected to the standby CD/CB Pump
1B and 1C Feedwater pumps are running

Which of the following occurs if the shaft shears between the reduction gear and the condensate pump casing for a running CD Pump?

A. 1CD152, CD pump recirc valve opens

B. 1CD157, GS condenser bypass valves A & B open

C. 1HD046A & B HDP discharge valves close

D. Both main feedwater pumps speeds decrease

Question #51

Given the following plant conditions:

Reactor power is 8%
A Feedwater isolation (FWI) occurred due to P-14
The startup feedwater pump is running

What actions MUST be performed in order to realign valves to establish main feedwater flow to the S/G's?

The P-14 signal must be _____

A. blocked and the main and aux FWI relays reset.

B. blocked and the reactor trip breakers need to be cycled open.

C. cleared and the FWI aux relays reset.

D. cleared, the reactor trip breakers cycled open, and main FWI relays reset.

Question #52

The diesel AFW pump has 2 battery packs each going to both starting motors with a selector switch determining which bank will power the starting motors. Each battery is designed to perform __(1)__ cranking cycles of __(2)__ secs each.

	(1)	(2)
A.	2	3
B.	3	4
C.	4	5
D.	5	6

Question #53

When 0RE-PR16J, 0A Blowdown After Filter Outlet Radiation Monitor, has a high radiation condition, the inlet valve to the Blowdown Monitor tank __(1)__ and the isolation valve to main condenser or CST __(2)__. The system is returned to normal __(3)__ after the radiation condition has cleared.

	(1)	(2)	(3)
A.	closes	opens	automatically
B.	opens	closes	manually
C.	closes	opens	manually
D.	opens	closes	automatically

Question #54

Waste gas decay tanks are designed to isolate at __(1)__ with a back up relief at __(2)__ .

	(1)	(2)
A.	80#	180#
B.	85#	170#
C.	90#	160#
D.	95#	150#

Question #55

Given the following plant conditions:

Unit 1 is in MODE 5
Unit 2 is in MODE 6
Main Control Room Ventilation radiation monitoring is provided by train A
Gas Monitor 0RE-PR032B, Control Room Gaseous Radiation Monitor, fails low

Which of the following is required to be performed?

A. Immediately, suspend all core alterations on Unit 2.

B. Within 1 hour initiate continuous monitoring using a portable monitor having the same alarm setpoint.

C. Within 1 hour, place the redundant Control Room Ventilation Filtration System in the normal mode.

D. Within 1 hour, shut down the Control Room Makeup System.

Question #56

Maintenance must be performed on a system that will require a CLEAN and a POTENTIALLY CONTAMINATED system to be aligned together through a temporary modification.

Which of the following is required to address the cross-contamination potential?

A. A manual isolation valve is required to be installed with a person stationed at the valve when it is open controlling flow.

B. The temporary modification crosstie shall have a caution card attached identifying the crosstie and potential of cross-contamination.

C. A check valve shall be installed in the temporary modification to prevent backflow between the two systems.

D. The temporary modification will have a relief valve installed in it to acuate at the clean systems operating pressure thereby preventing cross-contamination.

Question #57

Unit 2 is currently in MODE 4. At 0900 today, it is discovered that a routine 24-hour surveillance involving Shutdown Margin was last performed at 0600 on the previous day.

What is the required action in response to the failure to perform the surveillance?

A. The Technical Specification LCO 3.0.3 is applied.

B. The ACTION statement (LOCAR) is immediately initiated.

C. The surveillance may be delayed for up to 24 hours from the discovery per Technical Specification 4.0.3.

D. The surveillance requirements are satisfied if the surveillance is completed by 1200.

Question #58

The reactor was operating at 85% power with Control Bank D at 190 steps. Subsequently, a continuous rod withdrawal occurred followed by a turbine runback.

Which of the following is also expected for this condition?

A. Delta-I becomes more negative

B. DEHC MW IN Feedback light will be lit

C. TAVE CONT DEV HIGH will alarm

D. ROD BANK LOW INSERTION LIMIT alarm will be in

Question #59

The following conditions exist on Unit 1:

Reactor power 80%
Rod Deviation alarm lit
Rod Bottom alarm lit
Power Range Channel Deviation alarm lit
2 Rod Bottom LEDs lit on DRPI

Which of the following items describes the required operator response to this event?

A. Check Axial Flux Difference and Quadrant Power Tilt Ratio

B. Trip the reactor and perform 1BEP-0, "Reactor Trip or Safety Injection"

C. Restore rods per ROD-3, "Dropped or Misaligned Rod" then contact Nuclear Engineering to verify operability

D. Restore rods per ROD-3, "Dropped or Misaligned Rod" then verify operability by performing 1BOSR 1.4.2-1, Movable Control Assemblies Quarterly Surveillance

Question #60

Given the following plant conditions:

An inadvertent Unit 2 reactor trip occurred at 50% power
A loss of offsite power occurred when the Main Generator output breakers tripped
When the D/Gs energized the busses, an inadvertent SI occurred
All S/G NR levels have subsequently decreased to 38%

Which of the following describes operation of the AF Pumps under these conditions?

A. The 2A AF Pump is sequenced on after a time delay of 35 seconds and the 2B AF Pump started on RCP Bus Undervoltage.

B. The 2A AF Pump is sequenced on after a time delay of 35 seconds and the 2B AF Pump started due to low S/G levels.

C. The 2A AF Pump started due to low S/G levels when the D/G output breaker closed and the 2B AF Pump started on the SI signal.

D. The 2A AF Pump started due to low S/G levels when the D/G output breaker closed and the 2B AF Pump started on the loss of offsite power.

Question #61

Which of the following is the cause for a RAPID increase in Pressurizer level following a LOCA event with a loss of subcooling margin?

A. A PZR vapor space break.

B. SI accumulator Injection.

C. SI flow refilling the PZR.

D. PZR reference leg temperature decreased.

Question #62

A small break LOCA has occurred outside containment.

Actions of BCA-1.2 “LOCA Outside Containment”, have been completed and RCS pressure continued to decrease. A transition was made to BCA-1.1, “Loss of Emergency Coolant Recirculation”

Which of the following is the reason a transition was made to BCA-1.1?

A. To recover after the break was isolated

B. To terminate offsite release

C. To reverify that all automatic actions have been completed

D. To take compensatory actions for lack of inventory in the containment sump

Question #63

Which of the following describes the methods for depressurizing the RCS in preparation for Refill in the order of preference used in 1BEP ES-1.2, “Post LOCA Cooldown and Depressurization”?

A. One Pzr PORV
Normal Spray
Aux Spray

B. Normal Spray
One Pzr PORV
Aux Spray

C. Normal Spray
Aux Spray
Two Pzr PORVS

D. Two Pzr PORVS
Normal Spray
Aux Spray

Question #64

Which of the following will satisfy conditions necessary to MANUALLY OPEN Containment Recirculation Valve SI8811A?

1. SI8812A - open
2. SI8812A - closed

3. CS001A - open
4. CS001A - closed

5. RH8701A - open
6. RH8701B - closed

A. 1, 3, 5

B. 2, 3, 5

C. 1, 4, 6

D. 2, 4, 6

Question #65

A plant heatup was in progress in accordance with BGP 100-1, when a leak was detected by the actuation of alarm “CNMT DRAIN LEAK DETECT FLOW HIGH.”

Following stabilization of the leak rate, the following plant conditions exist:

PZR level 42% and stable
PZR pressure 1600 psig and stable
Charging flow is 98 gpm as read on FI-121
Total letdown flow is 75 gpm
Total seal injection flow is 27 gpm
RCP seal parameters are normal

Which of the following actions will identify the correct leak location?

A. Closing the RCS loop drain valves will isolate a tube leak in the excess letdown heat exchanger.

B. Closing the orifice isolation valves and the letdown line isolation valves will isolate the leak downstream of 1CV131 letdown line pressure control valve.

C. Closing the individual seal injection isolation MOVs will isolate the leak at the seal injection line flange to the RCPs seal package.

D. Closing the charging line CNMT isolation valves will isolate the leak at the discharge line from the in service regenerative heat exchanger.

Question #66

Given the following plant conditions:

Plant in Mode 5
RCS temperature is 195°F and stable
RCS pressure is 325 psig and stable
Train “A” RH is in service, Train “B” RH is inoperable (OOS for repairs)
RCS is intact
All systems aligned in normal configuration for present conditions

A loss of RH shutdown cooling occurs with the temperature rising, which of the following is the preferred method for heat removal in accordance with 1BOA PRI-10?

A. RWST gravity feed to RCS, spill through the PZR PORVS

B. SI Pump Hot Leg Injection with spill through the 2-inch vent.

C. Natural or forced RCS flow while steaming intact S/Gs.

D. Reflux cooling to any S/G with level equal to or greater the 27% NR level.

Question #67

The following plant conditions exist on Unit 2:

A load reduction from 32% power was initiated 5 minutes ago
Current reactor power is 28%
PZR pressure 2235 psig and stable
PZR level 30% and stable
S/G levels (NR) 37%A, 39%B, 37%C, 38%D and stable

If the 2D S/G level were to drop to 29% and then rise to 35% 20 seconds later, what would be the response of the ATWS Mitigating System (AMS)?

A. AMS actuation signal is generated; the reactor trips and the motor driven AF pump start.

B. AMS actuation signal is generated; the main turbine would trip and both AF pumps start.

C. AMS actuation signal is NOT generated because turbine power is below C-20 setpoint.

D. AMS actuation signal is NOT generated because of a time delay in the S/G level circuit.

Question #68

Unit 1 is at 100% power with the following plant conditions:

Increased ouput on variable heaters
Letdown flow is 75 gpm
Charging flow is 105 gpm
S/G levels are constant
Tavg/Tref are matched

Which of the following events is in progress?

A. The PZR level control channel has failed high.

B. An atmospheric steam dump valve has opened.

C. A S/G tube leak has occurred.

D. PZR spray bypass flow has increased.

Question #69

Given the following:

Reactor power is 100%. .
RCS Tavg is stable at 582°F on all 4 loops.
RCS pressure is stable at 2235 psig.
Containment Pressure is INCREASING.
Steam Flow on each SG is STABLE

1A SG Feed Flow is pegged HIGH
1A SG Main FW Reg Valve is full OPEN
1A SG pressure is STABLE
1A SG level is DECREASING

Which of the following events is in progress?

A. Feed Line Break INSIDE Containment.

B. Steam Line Break INSIDE Containment.

C. Main FW Reg Valve failed OPEN.

D. Feed Flow Indicator pegged HIGH.

Question #70

The Unit is operating at 18% power. Which of the following describes the status of the Reactor Coolant Pump breakers and Reactor Trip breaker if the bus frequency for ALL RCP's is 55 Hz for 1 second?

	RCP Breakers	Reactor Trip Breakers
A.	Open	Shut
B.	Shut	Open
C.	Open	Open
D.	Shut	Shut

Question #71

Given the following Unit 1 conditions:

Reactor Power is 6%
Startup FW pump is in service
A and B CD/CB pumps are running
Instrument Air pressure is at 70 psig and dropping due to a header leak

Loss of air to which of the following COMPONENTS would result in an automatic reactor trip?

A. Condensate Pump recirculation valve CD152

B. CVCS Charging Flow Control valve CV121

C. RCP #1 Seal Leakoff Isolation valve CV8141A

D. Main FW Reg Bypass valve FW510A

Question #72

The following stable conditions are encountered when surveying a room located in the auxiliary building RPA:

General Area Radiation level in room	60 mrem/hr
Radiation level at 30 cm from pipe	375 mrem/hr
Radiation level on contact with pipe elbow	400 mrem/hr
Contamination levels	850 dpm/cm2 beta-gamma
	0 dpm/cm2 alpha
Airborne radiation level	0.6 DAC

What are the correct radiological postings or labels required to reflect the current radiological conditions for this room?

A. "DANGER, HIGH RADIATION AREA"
"HOT ZONE"
"CAUTION, CONTAMINATED AREA".

B "CAUTION, RADIATION AREA"
"HOT ZONE"
"CAUTION, CONTAMINATED AREA".

C. "DANGER, HIGH RADIATION AREA"
"HOT SPOT"
"AIRBORNE RADIOACTIVITY AREA".

D. "CAUTION, RADIATION AREA"
"HOT SPOT"
"AIRBORNE RADIOACTIVITY AREA".

Question #73

An operator received radiation exposure at both Braidwood and Byron Stations during the year.

The exposure record until the last day of the year is:

	Braidwood	Byron
Deep Dose Equivalent (DDE)	275 mrem	75 mrem
Lens Dose Equivalent (LDE)	15 mrem	10 mrem
Committed Effective Dose Equivalent (CEDE)	120 mrem	25 mrem
Shallow dose Equivalent (SDE)	25 mrem	15 mrem
Committed Dose Equivalent (CDE)	25 mrem	5 mrem

On the last day of the year the individual, at Byron Station, was requested to work in an area where the known radiation rate is 280 mR/hr. The source of the radiation is a nearby HOT SPOT inside a pipe trap where crud has been collecting and it has been determined to be totally gamma radiation.

If the worker takes 15 minutes to complete the task, what is the individual's Total Effective Dose Equivalent (TEDE) for the year?

A. 450 mrem

B. 565 mrem

C. 595 mrem

D. 660 mrem

Question #74

Unit 1 was operating at 28% power when the Loop B Reactor Coolant Pump (RCP) tripped on overcurrent.

Which of the following describes the unit's initial response? (Assume NO operator action AND NO rod motion.)

A. A reactor trip occurs and unaffected loop Tave increases.

B. A reactor trip occurs and unaffected loop Tave decreases.

C. A reactor trip will NOT occur and unaffected loop Tave decreases.

D. A reactor trip will NOT occur and unaffected loop Tave increases.

Question #75

Given the following plant conditions:

A LOCA has occurred on Unit 1
Power has been lost to BUS 142
The crew is initiating 1BEP ES-1.3 “Transfer to Cold Leg Recirculation Alignment”

Which of the following describes the affect of the loss of this bus on the Cold Leg Recirculation Alignment?

The SI pump’s suction is supplied by ......

A. both “A” train RH and “B” train RH from redundant paths.

B. “A” train RH only via crosstie to the CV Pumps suction valve 1CV8804A.

C. “B” train RH only via 1B RH discharge supply.

D. “A” train RH only via crosstie to the SI pumps suction valve 1SI8804B.

Question #76

Given the following plant conditions:

Unit 1 Reactor power was 85% with all control systems in automatic
1A MFP tripped .
The operator initiated a turbine runback

What was the initial response of the PZR pressure control system during this event?

A. The PORVs were blocked from opening to maintain pressure above the low reactor trip setpoint.

B. The variable heaters and the backup heaters turn full on to raise pressure to normal.

C. PZR Spray valves will throttle open to reduce pressure to normal.

D. Both PZR PORVs open to maintain pressure below the high reactor trip setpoint.

Question #77

Pressurizer Level is programmed from auctioneered-high __(1)__. Without program level, the pressurizer volume is _(2)_ to accommodate reactor coolant system water volume changes while limiting pressure transients.?

	(1)	(2)
A.	Tave	insufficient
B.	Tave	adequate
C.	Tc	adequate
D.	Tc	insufficient

Question #78

The following rod position indications exist:

The DATA B failure light is lit
LED for 24 steps is lit

What will be the range of the rod, using the normal and maximum indication accuracies due to coil placement and thermal expansion?

A. 20-32

B. 18-30

C. 16-30

D. 14-28

Question #79

A small break LOCA occurred coincident with loss of offsite power. The actions of 1BEP ES-1.2 Post LOCA Cooldown and Depressurization, are being performed. Plant status is as follows:

1A CV pump and 1B SI pump are running
Pressurizer level is stable at 58%
RCS pressure is stable at 900 psig
RVLIS indicates that a void exists in the reactor vessel head

With RCS subcooling stable at 3°F, the operator turns on a set of backup heaters to raise subcooling margin.

Which of the following describes the expected RVLIS and pressurizer level response?

	RVLIS	Pressurizer
A.	Decrease	Increase
B.	Increase	Decrease
C.	Decrease	Decrease
D.	Increase	Increase

Question #80

The following plant conditions exist:

LOCA is in progress
Containment Spray actuated due to high containment pressure
Containment Spray signal has been reset
Actions of ES-1.3 “Transfer to Cold Leg Recirculation” have been completed
Containment pressure is now 17 psig

Offsite power is then lost and the D/G output breakers have just closed onto the ESF buses

How are the Containment Spray Pumps restarted?

A. The pumps will auto start 15 seconds following closure of the D/G output breakers.

B. The pumps will auto start 40 seconds following closure of the D/G output breakers.

C. The operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 15 seconds following closure of the D/G output breakers.

D. The operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 40 seconds following closure of the D/G output breakers.

Question #81

The normal containment purge system is DESIGNED to:

A. aid in temperature, pressure and humidity control during LOCA conditions.

B. backup the hydrogen recombiners and maintain the concentration <4% by volume.

C. reduce radiation levels in containment during normal reactor operation.

D. provide continuous access to containment after a planned reactor shutdown.

Question #82

If all fuel racks in the Spent Fuel Pool are filled with irradiated fuel assemblies, what is the MINIMUM boron concentration required to maintain a safe reactivity condition of less than 0.95 Keff? (Assume new Holtec SFP storage racks)

A. 0 ppm

B. 150 ppm

C. 300 ppm

D. 500 ppm

Question #83

What is the mechanism that MINIMIZES the effect of shrink on indicated narrow range level for the Unit 2 D-5 S/G's when a reactor trip occurs from 100% power?

A. The circulatory velocity in the downcomer increases causing a pressure decrease.

B. Constant tempering flow reduces the preheat requirements for the incoming feedwater.

C. The level program maintains mass constant in the S/G.

D. The location of the lower level tap experiences a rise in static pressure that tends to offset the drop in the steaming rate.

Question #84

A loss of the condenser air removal system has occurred. Which of the following is the expected sequence of alarms?

a. Condenser vacuum low, turbine trip, C-9 Bypass Permissive lights

b. C-9 Bypass Permissive lights, condenser vacuum low, turbine trip

c. Condenser vacuum low, C-9 Bypass Permissive lights, turbine trip

d. Turbine trip, C-9 Bypass Permissive lights, condenser vacuum low

Question #85

Which of the following conditions are required to MANUALLY close the SAT feed on a 6.9KV breaker?

a. No lockouts on SAT or UAT feed

b. All SAT trips are in

c. UAT Feed Brkr C/S in A/C

d. UAT Feed Brkr open

Question #86

Which of the following identifies ALL the Fire Protection Pumps that will be running if system water pressure falls to 128 psig?

A. Diesel Engine Fire Pump, Electric Motor Driven Fire Pump, and both Jockey Pumps (0A and 0B).

B. Electric Motor Driven Fire Pump and the 0B Jockey Pump.

C. Diesel Engine Fire Pump and the 0A Jockey Pump.

D. Electric Motor Driven Fire Pump and both Jockey Pumps (0A and 0B).

Question #87

The following plant conditions exist on Unit 2:

The “0" CC HX is in service with the 2A CC Pump running
CC Surge Tank level was at 55% and is now at 60%
“0" CC HX Radiation Monitor RE-PR009 HIGH radiation level alarm is in

Which of the following describes the response of the CC system for these conditions?

A. No automatic actions occur.

B. The CC Surge Tank Vent Valve 2CC017 will automatically close and 1CC017 remains open.

C. The CC Surge Tank will be automatically isolated from letdown, prior to the CC Surge Tank completely filling and pressurizing.

D. Both CC Surge Tank Vent Valves, 1/2CC017, will automatically close.

Question #88

The following conditions exist for a job to be performed on a system.

The general area radiation levels are 10 mrem/hr in the room.
The hot spot in the room is a pipe elbow that has radiation levels of 100 mrem/hr.
The job will be performed near the hot spot area.

(Assumptions: ALL 4 cases below have the same transition time to and from destinations. All shielding placement and removal is at 100 mrem/hr)

Choose the method that best reduces personnel exposure.

A. Two Radiation Control personnel hang and remove 1 tenth thickness of lead shielding on the hot spot in 1.5 hours for the job. The job is performed after the lead shielding is in place by using 2 operators for 3 hrs each on the job.

B. The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.

C. The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hrs.

D. The job is performed by using 2 operators for 3 hrs each on the job at the hot spot.

Question #89

Given the following conditions on Unit 1:

A LOCA has occurred.
The crew is in EP-0 at step 15 with the following plant conditions:

CETCs are reading 1090 °F
RCS pressure is 1950 psig
Containment pressure 6 psig and increasing
S/G pressures are 1180 psig
AFW maximum flow capability 400 gpm
S/G levels (NR): 1A S/G 25%, 1B S/G 24%, 1C S/G 26%, 1D S/G 30%

Based on the above conditions, what is the proper procedure to be in?

A. FR-C.1, “Response to Inadequate Core Cooling”

B. FR-H.1, “Response to a Loss of Secondary Heat Sink”

C. FR-Z.1, “Response to High Containment Pressure”

D. Transition to EP-1, “Loss of Reactor or Secondary Coolant”

Question #90

Unit 1 is at 100% power. Which of the following describes the plant response if the controlling pressurizer level channel fails HIGH with NO operator action taken?

A. The PZR heaters trip and letdown isolates on low level. The reactor eventually trips on actual high PZR level.

B. PZR level decreases until the reactor trips on low pressure. Letdown then isolates when level drops to 17%.

C. PZR level decreases initially, but stabilizes below the programmed setpoint. The controller will then restore level to program with an appropriate time constant.

D. The PZR heaters trip and letdown isolates on low level. The PZR will then gradually fill until a high pressure reactor trip occurs.

Question #91

The plant was operating at 10% Reactor Power when a loss of offsite power caused the RCPs to trip. Identify ALL of the indications that verify natural circulation is occurring.

1 - Core exit thermocouples --- decreasing
2 - Core exit thermocouples --- stable or increasing

3 - RCS hot leg temperature --- stable or decreasing
4 - RCS hot leg temperature --- increasing

5 - RCS subcooling --- decreasing
6 - RCS subcooling --- increasing

7 - RCS cold leg temperature --- at saturation for SG pressure
8 - RCS hot leg temperature --- at saturation for SG pressure

A. 1, 4, 5, 7

B. 2, 4, 6, 8

C. 1, 3, 6, 7

D. 2, 3, 5, 8

Question #92

Which of the following statements explains the BEP-1, "Loss of Reactor or Secondary Coolant," bases for stopping the RCPs as directed by the Operator Action Summary page following a containment Phase B actuation?

A. Delays the onset of two phase flow.

B. Preempt the RCP's tripping on cavitation because it is assumed that if containment spray actuates, an RCS depressurization is in progress.

C. Reduces the containment high pressure transient by lowering the energy release rate to containment from forced flow.

D. Precludes RCP bearings and seals from overheating on loss of component cooling water.

Question #93

Which of the following list ALL administrative requirements and interlocks associated with opening cold leg recirculation valves SI8811A and SI8811B.

A. No SI signal present
RWST level 45%
4 sump lights lit for RHR Pump NPSH.

B. SI signal present
RWST level 45%
2 sump lights lit for RHR Pump NPSH.

C. No SI signal present
RWST level 46%
2 sump lights lit for RHR Pump NPSH.

D. SI signal present
RWST level 46%
4 sump lights lit for RHR Pump NPSH.

Question #94

If the Reactor Coolant Subcooling Margin Monitor is not working properly, how will the subcooling margin be calculated?

A. Use 5 highest CETC average and RCS wide range pressure to determine subcooling margin.

B. Use 5 lowest CETC average and RCS wide range pressure to determine subcooling margin.

C. Use 10 highest CETC average and RCS wide range pressure to determine subcooling margin.

D. Use 10 lowest CETC average and RCS wide range pressure to determine subcooling margin.

Question #95

Which of the following is a positive indication that the PRT has ruptured following a pressurizer PORV failing full OPEN?

A. PRT temperature is decreasing.

B. PORV relief line temperature is increasing.

C. PRT level decreases to its normal value of 70%.

D. Pressurizer level is decreasing.

Question #96

Both units are at 100% power. The Component Cooling (CC) system is in its alignment for normal operations with ALL equipment operable.

A leak occurs resulting in the following conditions on Unit 2:

Alarm window for CC SURGE TANK LEVEL HIGH LOW actuates.
CC Surge Tank level is 33% and slowly falling
Demin Water and Primary Water makeup valves indicate OPEN
RCS temperature (average Tave) is 582°F and stable
PZR level is 60% and stable
VCT level is 42% and stable
Charging and letdown flows are balanced and normal Spent Fuel Pool level is stable

Where is the location of the CC System leak?

A. The seal water heat exchanger

B. The 2A RH pump seal cooler

C. The 2B letdown heat exchanger

D. The 2B excess letdown heat exchanger

Question #97

Which of the following is an indication that recombination is occurring after having placed the Hydrogen Recombiners in service?

A. Hydrogen Recombiner power increases to 20 KW.

B. Containment dewpoint decreases after Hydrogen Recombiners are placed in service.

C. Hydrogen Recombiner average thermocouple temperature is at or above 1200°F.

D. Containment pressure ~~deceases~~ *decreases* after Hydrogen Recombiners are placed in service.

Question #98

Which of the following is a function of the Service Air system?

A. to supply air to the Instrument Air system

B. to supply air to the primary emergency breathing air system

C. to supply air to only essential components

D. to supply oil filled compressed air for maintenance use

Question #99

During a cooldown on Unit 1 the following conditions exist:

RCS loop Tave: Loop 1: 550°F decreasing
Loop 2: 548°F decreasing
Loop 3: 551°F decreasing
Loop 4: 548°F decreasing
Steam header pressure- 1030 psig and decreasing
Steam Dump Mode Selector switch-STM PRESS MODE
Steam Dump Controller-MAN set at 30% demand

The operator momentarily places the Train A and Train B Steam Dump Bypass Interlock switches to Bypass and then releases them.

What is the status of the Steam Dump valves following the operator's actions?

A. All valves are fully closed

B. Three valves in group 1 are partially open

C. Three valves in group 1 are fully open and valves in group 2 are fully shut.

D. Three valves in group 1 are fully open and three valves in group 2 are partially open.

Question #100

The primary basis for depressurizing all intact steam generators to atmospheric pressure in FR-C.1, "RESPONSE TO INADEQUATE CORE COOLING," is to:

A. insure core exit thermocouple temperatures are reduced to less than 700 °F.

B. reduce S/G pressure to increase feedwater flow.

C. reduce RCS pressure for establishing low-head safety injection.

D. enhance natural circulation cooling of the reactor core.

Enclosure 5

WRITTEN EXAMINATION AND ANSWER KEYS (SRO)

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / $\boxed{\text{III}}$ / IV
Date: June 29, 2000	Facility/Unit:
License Level: RO / SRO	Reactor Type: $\boxed{\text{W}}$ / CE / BW / GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

Byron SRO Written Examination
Answer Key

1	D	26	C	51	C	76	A
2	C	27	C	52	A	77	B
3	A	28	B	53	D	78	A
4	B	29	D	54	B	79	D
5	D	30	C	55	C	80	D
6	A	31	C	56	D	81	B
7	C	32	D	57	C	82	B
8	D	33	C	58	D	83	B
9	A	34	B	59	C	84	B
10	B	35	A	60	A	85	D
11	D	36	A	61	D	86	D
12	C	37	D	62	D	87	C
13	C	38	B	63	B	88	B
14	B	39	D	64	B	89	C
15	D	40	D	65	A	90	B
16	C	41	C	66	C	91	C
17	A	42	D	67	D	92	A
18	B	43	C	68	D	93	C
19	B	44	A	69	C	94	B
20	C	45	C	70	A	95	B
21	D	46	D	71	B	96	D
22	A	47	C	72	C	97	C
23	D	48	B	73	A	98	B
24	A	49	~~C~~ D	74	C	99	B
25	B	50	B	75	C	100	D

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
1	D	000005K3.01	4.3	3	H	New	ES-0.1 Step 5
2	C	000015A1.22	4.2	4	H	Bank	1 BOA RCP-1
3	A	W/E09K2.02	3.9	3	H	Bank	LOCA Procedure Lesson Plan
4	B	000026A2.01	3.5	4	H	Bank	BOA PRI-6 Att A CC Lesson Plan
5	D	000027A1.01	3.9	3	H	New	BAR 1-12-A1
6	A	000040K3.02	4.4	3	H	Bank	SSPS Lesson Plan
7	C	000051A2.02	4.1	3	H	New	1 BOA SEC-3
8	D	000055G2.1.10	3.9	2	F	New	ECA 0.0 Lesson Plan
9	A	000057G2.4.10	3.1	3	H	Bank	1BOP ELEC-2
10	B	000062A1.01	3.1	2	H	New	Containment Ventilation Lesson Plan ESW Lesson Plan 1BOA PRI-7
11	D	013A4.02	4.3	3	F	Bank	FW-1 FW Lesson Plan
12	C	000068A2.09	4.3	3	H	New	Steam Tables
13	C	061K1.02	3.7	3	F	New	AFW Lesson Plan
14	B	000074K1.03	4.9	3	F	Bank	1 BFR-C.1 Procedure Lesson Plan
15	D	G2.1.1	3.8	2	F	New	BAP 300-1 Conduct of Ops
16	C	G2.1.3	3.4	2	F	Bank	OP-AA-101-401
17	A	G2.1.12	4.0	3	H	New	ITS 3.4.13
18	B	001K5.05	3.9	4	H	New	1BCB-1 Tables
19	B	003A1.09	2.8	3	H	Bank	RCP Lesson Plan
20	C	004A4.18	4.1	3	F	New	CVCS Lesson Plan ITS Boration flow path
21	D	013A3.02	4.2	3	H	Bank	EF-2 ESF setpoint
22	A	000038K3.08	4.5	2	F	Bank	1BEP-3
23	D	017K6.01	3.0	2	F	Bank	Incore Instrumentation Lesson Plan
24	A	022G2.1.32	3.8	2	F	Bank	ITS 3.6.5

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
25	B	056A2.04	2.8	4	H	New	MFW Lesson Plan
26	C	059K4.19	3.4	3	H	Bank	MFW Lesson Plan
27	C	061K2.03	3.8	2	F	New	AF-1 AFW Lesson Plan
28	B	068K4.01	4.1	2	F	New	Liquid Rad Waste Lesson Plan
29	D	071K4.01	3.0	2	F	New	RW-1 Gaseous Radwaste Lesson Plan
30	C	072G2.1.14	3.3	2	H	Bank	Control Room HVAC Lesson Plan ITS 3.3.6
31	C	G2.2.11	3.4	2	F	New	CC-AA-112, Temp Mods
32	D	G2.2.12	3.4	2	F	Bank	ITS SR 3.0.2
33	C	000001A2.05	4.6	3	H	Bank	1BOA ROD-1
34	B	000003G2.2.4	3.0	2	H	New	1BOA ROD-3
35	A	064K4.11	3.9	3	H	Bank	AFW Lesson Plan EF-1 ESF setpoints
36	A	000008A2.12	3.7	3	F	New	PZR Lesson Plan
37	D	W/E04K1.02	4.2	2	F	New	1 BCA-1.1
38	B	W/E03K3.03	3.9	2	F	New	1BEP ES-1.2
39	D	W/E11K2.01	3.9	2	F	New	ECCS Lesson Plan ECCS-3
40	D	000022A2.02	3.2	3	H	Bank	1 BOA PRI-1
41	C	000025K3.01	3.4	3	H	New	1 BOA PRI-10
42	D	000029A1.15	3.9	3	H	New	AMS Lesson Plan
43	C	000037A2.01	3.4	4	H	Bank	1 BOA SEC-8
44	A	000054K1.01	4.3	3	H	New	1BEP-2 Background info EP-2
45	C	000007K2.02	2.8	2	H	New	BAR 1-11-B5
46	D	078K3.02	3.2	3	H	Bank	Instrument Air Lesson Plan
47	C	G2.3.1	3.0	3	H	Bank	NGET
48	B	G2.3.4	3.1	3	H	Bank	NGET

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
49	~~C~~ D	002K6.02	3.8	2	F	New	ESF Setpoint EF-1 Figure HT-7-22C
50	B	006K2.04	3.8	3	H	New	ECCS Lesson Plan
51	C	010A3.02	3.5	3	H	New	PZR Lesson Plan
52	A	011K5.12	3.3	2	F	Bank	PZR Lesson Plan
53	D	014K5.01	3.0	4	H	New	Rod Position Lesson Plan TS bases
54	B	016K3.02	3.53.5	3	H	New	BFR-I.3 Lesson Plan BEP ES-0.2 BEP ES 1.2
55	C	026A4.01	4.1	3	H	Bank	CS Lesson Plan
56	D	029G2.1.27	2.9	2	F	New	VP-2 Containment Purge
57	C	033K4.05	3.3	2	F	New	TS 3.7.15
58	D	035K5.03	3.8	3	F	Bank	SG Lesson Plan
59	C	055K3.01	2.5	2	F	New	SEC-3 Condenser Vacuum
60	A	062K2.01	3.4	2	F	New	AC Power Lesson Plan
61	D	086A2.02	3.3	2	F	Bank	Fire Protection Lesson Plan
62	D	073K1.01	3.9	3	H	New	Component Cooling Lesson Plan
63	B	G2.3.10	3.3	3	H	New	NGET
64	B	G2.4.1	4.6	2	H	New	1BFR-H.1
65	A	00028K2.03	2.9	3	H	New	PZR Lesson Plan
66	C	000056K1.01	4.2	2	F	Bank	1BCA-0.1
67	D	000011G2.4.18	3.6	2	F	New	1 BEP-1 fold out page
68	D	G2.4.2	4.1	2	F	Bank	1BEP ES1.3 ESF Lesson Plan
69	C	G2.4.3	3.8	4	F	New	ITS 3.3.i
70	A	007A2.01	4.2	2	F	New	1BEP-0
71	B	008K3.01	3.5	4	H	New	1BOA PRI-6 Att A
72	C	028A4.01	4.0	3	F	New	0 BOSR 6.8.1-1

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
73	A	079G2.1.28	3.3	2	F	New	Service Air Lesson Plan
74	C	041K4.09	3.3	3	H	Bank	Steam Dump Lesson Plan
75	C	G2.4.6	4.0	2	F	New	FR-C1 Procedure Lesson Plan
76	A	015A1.04	3.7	2	F	New	QPTR SR 3.2.4.2
77	B	034K4.02	3.3	3	F	New	Fuel Handling Lesson Plan
78	A	W/E08K1.2	4.0	2	F	New	1BFR-C2 FRC Lesson Plan
79	D	W/E01K2.2	3.9	3	H	New	1BCA-1.1 ECCS Lesson Plan
80	D	000024K3.01	4.4	3	H	New	1 BOA PRI-2 Lesson Plan RMC Lesson Plan
81	B	061000A104	3.9	3	H	Bank	TS Bases 3/4.7.6 AFW Lesson Plan
82	B	000055K3.02	4.6	2	H	New	BCA-0.0 Lesson Plan
83	B	000067AA2.17	4.3	2	F	New	2BOA PRI-5 Lesson Plan
84	B	E/03K2.01	4.0	2	F	New	1 BEP-1
85	D	000076A2.02	3.8	2	F	Bank	1BOA-PRI4 TS 3.4.16
86	D	000036A2.02	4.1	2	H	New	BAR 1-6-C3
87	C	G2.1.11	3.8	2	F	New	TS 3.4.2
88	B	G2.2.23	3.8	3	H	New	TS 3.8.1
89	C	000033G2.1.12	4.0	3	H	New	TS 3.3.1
90	B	000038EA2.01	4.5	4	H	New	1BEP-3
91	C	0750000A2.03	2.7	3	H	Mod	Steam Dump Lesson Plan Circ Water Lesson Plan
92	A	W/E03K3.03	3.9	3	F	New	1 BFR-I.3
93	C	000060AK2.01	2.9	2	F	New	Fuel Handling Lesson Plan
94	B	026000G2.4.20	4.0	2	F	New	Lesson Plan Z.1-3RI.doc
95	B	G2.1.10	3.9	2	F	New	TS 3.4.16 bases
96	D	G2.4.45	3.6	2	F	New	Lesson Plan S49R01.doc

Q#	Ans	K/A	Imp. Rating	LOD	LOK	Source	Reference
97	C	W/E14EK1.04	3.6	3	H	New	1BFR-P.1 1BFR-H.1
98	B	000007A2.04	2.8	3	F	Bank	BFR-S.1, S.2, S.3
99	B	000009A2.34	4.2	3	H	Bank	1BEP-1, Fig 1 BEP 1-1
100	D	G2.2.6	3.3	2	F	New	AD-AA-101

Question #100

The transition is made from EP-0 to ES-0.1 on Unit 1. Step 4 in ES-0.1 requires boration for all rods NOT fully inserted. There are 3 rods not fully inserted into the core at this point. What is the MINIMUM gallons that will have to be borated from the RWST for the three rods?

A. 1320 gallons

B. 5500 gallons

C. 3960 gallons

D. 16500 gallons

Question #101

Given the following conditions:

Unit 1 is operating at 100% power
RCP No. 1 SEAL LEAKOFF FLOW HIGH alarm is received
No. 2 seal leakoff high flow alarm has been printed
RCP No. 1 seal leakoff recorder indication is high offscale on the high range
Make-up to the RCS has increased 40 gpm to maintain PZR level

Which of the following has occurred and what action is required?

A. The No. 1 and No. 2 seals have failed and a controlled reactor shutdown is required.

B. Only the No. 2 seal has failed and continued monitoring of RCP conditions is required.

C. The No. 1 seal has failed and an immediate reactor trip is required.

D. The No. 2 and No. 3 seals have failed and continued monitoring of RCP conditions is required.

Question #102

During a small break LOCA on a cold leg, a phase is reached where the vessel level continues to decrease below the hot leg penetrations and boiling in the core is the means of transporting the core heat to the bubble. A fixed differential pressure exists between the core and the break and is maintained by the loop seal.

What is the primary mechanism for heat removal during this phase?

A. Condensation of vapor from the bubble at the hot leg side of the SG U-tubes which then drains back to the core via the hot legs.

B. Condensation of vapor in the head, which is cooled by fans in containment, and draining back to the core.

C. Slug flow via the cold legs through the loop seal and flashing across the cold leg break.

D. Partial natural circulation flow characterized by liquid pulses flowing from the cold leg over the U-tubes and into the hot legs.

Question #103

The following plant conditions exist:

The reactor is shutdown
RCS temperature is 290°F and stable
RCS pressure is 320 psig and stable
RH is in shutdown cooling
RH Letdown is in service
CC surge tank level is slowly decreasing with the makeup valves to CC surge tank fully open

A leak has occurred in the ?

A. RH Heat Exchanger

B. Seal Water Heat Exchanger

C. Letdown Heat Exchanger

D. Thermal Barrier Heat Exchanger

Question #104

Given the following conditions on Unit 1:

Reactor power is steady at 100%
Tave is steady at 582°F
PZR level is 60% and slightly increasing
PZR pressure is 2230 psig and slowly decreasing
ALL systems are aligned normally

Which of the following conditions has occurred?

A. LK-459 PZR level controller has failed high

B. PZR PORV 456 is full open

C. PZR pressure transmitter PT-458 has failed high

D. PZR spray valve RY455B has failed to 50% open

Question #105

Unit 1 has tripped due to a steamline break inside containment. Shortly after the trip, the following parameters were recorded:

PZR pressure 1750 psig and stable
PZR level 22% and stable
CNMT pressure 7.8 psig (on all instruments)

S/G level(NR)	31% A,	30% B	25% C	34% D
S/G pressures	760 psig 1A	775 psig 1B	680 psig 1C	800 psig 1D

A steamline isolation occurred due to ....?

A. S/G low pressure steamline isolation.

B. S/G pressure rate steamline isolation.

C. the containment pressure circuit for steamline.

D. the PZR low pressure SI.

Question #106

The plant has the following conditions:

Reactor Power 52% steady state
Generator load is steady at 600MW
Condenser vacuum 2.2 in.HgA and steady

A leak developed in one of the water boxes causing pressure to rise at the rate of 0.2 inches HgA/minute. After 2 minutes, the operator began a load decrease at the rate of 10MW/minute in an attempt to offset the pressure rise and reduce load below the P-8 setpoint.

Assuming the load decrease remained constant and the rate of pressure rise remained constant throughout the event, what action is required? (References are provided)

A. The operator would initiate a manual turbine trip after the load is reduced to less than 30%.

B. No operator action, the turbine will automatically trip at 35% power causing a reactor trip.

C. The operator will initiate a manual reactor trip at approximately 39% power.

D. The operator will initiate a manual reactor trip at approximately 47% power.

Question #107

Byron is considered an Alternate AC (AAC) Station for design analysis during a Station Blackout. Which of the following is a reason that Byron chose to qualify as an AAC station instead of a 4-hour coping station?

A. EDG’s have a high availability rate and are 100% redundant

B. AAC source is available within 30 minutes.

C. AAC source is capable of carrying the immediate needs of both units for 8 hours

D. Capable of unit cross-tying of auxiliary power from the Main Control Room.

Question #108

An operator noted the following annunciators were in following an event: (Not all alarms are provided)

PWR RNG HIGH STPT RX TRIP ALERT
OPDT HIGH ROD STOP C-4
OTDT HIGH ROD STOP C-3
PZR PRESS CONT DEV LOW HTRS ON
RCP BUS UNDERVOLT RX TRIP ALERT
RCP 1C BRKR OPEN OR FLOW LOW ALERT
TURB STOP VLV CLOSED ALERT

Which bus lost power?

A. Instrument Bus 113

B. Instrument Bus 112

C. DC Bus 113

D. DC Bus 112

Question #109

Which of the following describes the effect on containment if the Essential Service Water supply to ALL Reactor Containment Fan Coolers (RCFC) is secured? (Assume normal 100% power operation.)

Containment temperature would...

A. remain the same since the other containment HVAC equipment would maintain cooling.

B. increase since Containment Chiller will also trip upon Essential Service Water isolation.

C. increase because only Essential Service Water supplies RCFC's.

D. increase since Component Cooling can only supply RCFC's with a manual lineup.

Question #110

The plant was operating at 50% power when an inadvertent safety injection occurred. The operators wish to regain control of feedwater valves in order to feed the steam generators using the startup feedwater pump.

Which of the following is the correct order of actions that will be successful in restoring control of feedwater valves:

A. Reset SI, reset FW Isolation, cycle reactor trip breakers, reset FW Isolation Aux relays

B. Reset FW Isolation, reset SI, cycle reactor trip breakers, reset FW Isolation Aux relays

C. Reset SI, cycle reactor trip breakers, reset FW Isolation Aux relays, reset FW Isolation.

D. Reset SI, cycle reactor trip breakers, reset FW Isolation, reset FW Isolation Aux relays.

Question #111

The Control Room has been evacuated in accordance with BOA PRI-5 and the operators are performing an RCS cooldown.

The plant conditions are as follows:

Reactor coolant temperature is 456°F and stable
Reactor coolant pressure is 449 psig and stable

Which of the following describes the approximate state of the Reactor coolant when checking subcooling margin? (References are provided)

It is about ....?

A. 3 degrees superheated

B. at the saturation point

C. 3 degrees subcooled

D. 12 degrees subcooled

Question #112

Which of the following describes the relationship between the Unit 1 Auxiliary Feedwater System (AFW) piping and the Main Feedwater System piping?

The AFW piping connects downstream of the...

A. MFW bypass valves 6-inch piping and upstream of the FWIV FW035A.

B. MFW regulating valves 14-inch piping and upstream of the FWIV FW034A.

C. FWIV FW035A and upstream of the containment penetration.

D. Containment penetration and upstream of the last feedwater check valve after FWIV FW034A and prior to the SG.

Question #113

Which of the following sets of actions states the proper sequence of major actions to be performed in accordance with 1BFR-C.1, "Response to Inadequate Core Cooling", for removing heat from the core?

A. Restoration of ECCS flow
RCP restart
Rapid secondary depressurization

B. Restoration of ECCS flow
Rapid secondary depressurization
RCP restart

C. RCP restart
Restoration of ECCS flow
Rapid secondary depressurization

D. RCP restart
Rapid secondary depressurization
Restoration of ECCS flow

Question #114

A non-licensed individual may move control rods using the IN/HOLD/OUT switch located in the control room under which of the following conditions?

The non-licensed individual is ...

B. a plant operator performing a surveillance test and is directly supervised by the on shift NSO.

C. a qualified nuclear engineer performing a control rod test and is directly supervised by a previously licensed NSO for that unit.

D. a plant operator who is enrolled in the initial license training program and is directly supervised by a certified instructor of the class.

E. a maintenance manager who is enrolled in the initial license training program and is under the direct supervision of the on shift NSO.

Question #115

An NRC-licensed operator works shift Monday morning as an NSO for 8 hours on Unit 1. The same individual is off work on Tuesday. On Wednesday morning the same operator stands the Unit 1 NSO watch for 8 hours. The same individual is off of work on Thursday. On Friday night the same operator is assuming the Unit 1 NSO watch at shift turnover

What is the administrative procedural requirement associated with reviewing the Unit logs?

A. Thursday only.

B. Thursday and Friday only.

C. Wednesday, Thursday, and Friday only.

D. A minimum of the past five days.

Question #116

Given the following conditions on Unit 2:

Reactor Power is 100%

A leak rate surveillance indicates the following:
Total RCS leakage rate is 9.0 gpm
Leakage to PRT is 6.0 gpm
Leakage to Reactor Coolant Drain Tank is 2.0 gpm
Leakage into Secondary from Primary as follows:

Unit 2 A S/G .07 GPM
B S/G .08 GPM
C S/G .09 GPM
D S/G .10 GPM

Which of the following statements are correct concerning the above conditions?

A. No leakage limits have been exceeded.

B. Unidentified leakage limit has been exceeded.

C. Total Primary to Secondary leakage limit has been exceeded.

D. Secondary leakage limit through one S/G has been exceeded.

Question #117

Which of the following operations results in the largest reactivity change? (References are provided)

A. Inserting 10 steps with rods initially at 190 steps on CBD at 100% power at 50 EFPH.

B. Inserting 10 steps with rods initially at 190 steps on CBD at 0% power at 11,500 EFPH

C. Withdrawing 10 steps with rods initially at 190 steps on CBD at 100% power at 11,500 EFPH.

D. Withdrawing 10 steps with rods initially at 190 steps on CBD at 0% power at 50 EFPH

Question #118

How would the RCP seals be affected if 1CV8142, #1 Seal Bypass Valve, was opened with the associated RCP running at normal operating pressure in RCS?

A. Flow across the #1 seal will fall to 0 gpm and the seal will be damaged by overheating.

B. Differential pressure changes across the #1 seal resulting in unbalanced seal motion.

C. Full RCS pressure is applied to the #3 Seal causing it to become the primary seal.

D. Pressure to the seal return line to the VCT is lowered causing flow across #2 seal to drop.

Question #119

Why is the manual emergency boration valve, CV8439, not used for performing emergency boration?

A. There is no way to monitor flow through the valve when in use so total boration flow could not be determined.

B. The throttling characteristics of the valve are poor, thereby resulting in full flow of 75 gpm or no flow at all.

C. The valve will only allow 10 gpm flow thereby not meeting the criteria for emergency boration.

D. Locally operated valves are not analyzed for safety functions and thereby not considered for performing safety function.

Question #120

Given the following plant conditions on Unit 1:

Reactor power was at 100% when a spurious SI signal was generated
Reactor Trip Breaker B failed to open
The spurious SI signal was cleared
The RH pumps, SI pumps, and 1A CV pump were secured.

After the ECCS pumps were secured, a small break LOCA occurred.

Which of the following occurs when containment pressure rises to 10 psig? (Assuming no operator actions are taken)

A. Only the MSIV and MSIV bypass valves close.

B. 1B and 1C MSIV’s close but the 1A and 1D MSIV’s remain open.

C. The 1A RH, 1A SI, and 1A CV Pumps start; the MSIV and MSIV bypass valves close.

D. The 1B RH and 1B SI Pumps start; the MSIV and MSIV bypass valves close.

Question #121

Which of the following determines the target temperature at which RCS cooldown is terminated following a S/G tube rupture using 1BEP-3, “Steam Generator Tube Rupture”?

A. The ruptured S/G pressure.

B. RCS subcooling of 39 degrees.

C. The lowest intact S/G pressure.

D. Maximum temperature for placing RH in service in the event of a loss of High Head Flow.

Question #122

A LOCA has occurred. Core exit thermocouple temperatures are indicating 690°F and increasing rapidly.

The Incore Thermocouples are providing satisfactory indication and will become __(1)__ accurate above __(2)__. (Assume NO core cooling is present)

	(1)	(2)
A.	less	700 °F
B.	more	1800 °F
C.	more	700 °F
D.	less	1800 °F

Question #123

The containment average temperature is the calculated average of the RCFC Dry Bulb _____.

A. inlet temperature of those RCFC's that are running.

B. outlet temperature of all RCFC's regardless of operating status.

C. inlet temperature of all RCFC's regardless of operating status.

D. outlet temperature of those RCFC's that are running.

Question #124

Given the following plant conditions on Unit 1:

Reactor power is 100%
3 CD/CB pumps are running
CD/CB Pump Selector Position is selected to the standby CD/CB Pump
1B and 1C Feedwater pumps are running

Which of the following occurs if the shaft shears between the reduction gear and the condensate pump casing for a running CD Pump?

A. 1CD152, CD pump recirc valve opens

B. 1CD157, GS condenser bypass valves A & B open

C. 1HD046A & B HDP discharge valves close

D. Both main feedwater pumps speeds decrease

Question #125

Given the following plant conditions:

Reactor power is 8%
A Feedwater isolation (FWI) occurred due to P-14
The startup feedwater pump is running

What actions MUST be performed in order to realign valves to establish main feedwater flow to the S/G's?

The P-14 signal must be _____

A. blocked and the main and aux FWI relays reset.

B. blocked and the reactor trip breakers need to be cycled open.

C. cleared and the FWI aux relays reset.

D. cleared, the reactor trip breakers cycled open, and main FWI relays reset.

Question #126

The diesel AFW pump has 2 battery packs each going to both starting motors with a selector switch determining which bank will power the starting motors. Each battery is designed to perform __(1)__ cranking cycles of __(2)__ secs each.

	(1)	(2)
A.	2	3
B.	3	4
C.	4	5
D.	5	6

Question #127

When 0RE-PR16J, 0A Blowdown After Filter Outlet Radiation Monitor, has a high radiation condition, the inlet valve to the Blowdown Monitor tank __(1)__ and the isolation valve to main condenser or CST __(2)__. The system is returned to normal __(3)__ after the radiation condition has cleared.

	(1)	(2)	(3)
A.	closes	opens	automatically
B.	opens	closes	manually
C.	closes	opens	manually
D.	opens	closes	automatically

Question #128

Waste gas decay tanks are designed to isolate at __(1)__ with a back up relief at __(2)__ .

	(1)	(2)
A.	80#	180#
B.	85#	170#
C.	90#	160#
D.	95#	150#

Question #129

Given the following plant conditions:

Unit 1 is in MODE 5
Unit 2 is in MODE 6
Main Control Room Ventilation radiation monitoring is provided by train A
Gas Monitor 0RE-PR032B, Control Room Gaseous Radiation Monitor, fails low

Which of the following is required to be performed?

A. Immediately, suspend all core alterations on Unit 2.

B. Within 1 hour initiate continuous monitoring using a portable monitor having the same alarm setpoint.

C. Within 1 hour, place the redundant Control Room Ventilation Filtration System in the normal mode.

D. Within 1 hour, shut down the Control Room Makeup System.

Question #130

Maintenance must be performed on a system that will require a CLEAN and a POTENTIALLY CONTAMINATED system to be aligned together through a temporary modification.

Which of the following is required to address the cross-contamination potential?

A. A manual isolation valve is required to be installed with a person stationed at the valve when it is open controlling flow.

B. The temporary modification crosstie shall have a caution card attached identifying the crosstie and potential of cross-contamination.

C. A check valve shall be installed in the temporary modification to prevent backflow between the two systems.

D. The temporary modification will have a relief valve installed in it to acuate at the clean systems operating pressure thereby preventing cross-contamination.

Question #131

Unit 2 is currently in MODE 4. At 0900 today, it is discovered that a routine 24-hour surveillance involving Shutdown Margin was last performed at 0600 on the previous day.

What is the required action in response to the failure to perform the surveillance?

A. The Technical Specification LCO 3.0.3 is applied.

B. The ACTION statement (LOCAR) is immediately initiated.

C. The surveillance may be delayed for up to 24 hours from the discovery per Technical Specification 4.0.3.

D. The surveillance requirements are satisfied if the surveillance is completed by 1200.

Question #132

The reactor was operating at 85% power with Control Bank D at 190 steps. Subsequently, a continuous rod withdrawal occurred followed by a turbine runback.

Which of the following is also expected for this condition?

A. Delta-I becomes more negative

B. DEHC MW IN Feedback light will be lit

C. TAVE CONT DEV HIGH will alarm

D. ROD BANK LOW INSERTION LIMIT alarm will be in

Question #133

The following conditions exist on Unit 1:

Reactor power 80%
Rod Deviation alarm lit
Rod Bottom alarm lit
Power Range Channel Deviation alarm lit
2 Rod Bottom LEDs lit on DRPI

Which of the following items describes the required operator response to this event?

A. Check Axial Flux Difference and Quadrant Power Tilt Ratio

B. Trip the reactor and perform 1BEP-0, "Reactor Trip or Safety Injection"

C. Restore rods per ROD-3, "Dropped or Misaligned Rod" then contact Nuclear Engineering to verify operability

D. Restore rods per ROD-3, "Dropped or Misaligned Rod" then verify operability by performing 1BOSR 1.4.2-1, Movable Control Assemblies Quarterly Surveillance

Question #134

Given the following plant conditions:

An inadvertent Unit 2 reactor trip occurred at 50% power
A loss of offsite power occurred when the Main Generator output breakers tripped
When the D/Gs energized the busses, an inadvertent SI occurred
All S/G NR levels have subsequently decreased to 38%

Which of the following describes operation of the AF Pumps under these conditions?

A. The 2A AF Pump is sequenced on after a time delay of 35 seconds and the 2B AF Pump started on RCP Bus Undervoltage.

B. The 2A AF Pump is sequenced on after a time delay of 35 seconds and the 2B AF Pump started due to low S/G levels.

C. The 2A AF Pump started due to low S/G levels when the D/G output breaker closed and the 2B AF Pump started on the SI signal.

D. The 2A AF Pump started due to low S/G levels when the D/G output breaker closed and the 2B AF Pump started on the loss of offsite power.

Question #135

Which of the following is the cause for a RAPID increase in Pressurizer level following a LOCA event with a loss of subcooling margin?

A. A PZR vapor space break.

B. SI accumulator Injection.

C. SI flow refilling the PZR.

D. PZR reference leg temperature decreased.

Question #136

A small break LOCA has occurred outside containment.

Actions of BCA-1.2 “LOCA Outside Containment”, have been completed and RCS pressure continued to decrease. A transition was made to BCA-1.1, “Loss of Emergency Coolant Recirculation”

Which of the following is the reason a transition was made to BCA-1.1?

A. To recover after the break was isolated

B. To terminate offsite release

C. To reverify that all automatic actions have been completed

D. To take compensatory actions for lack of inventory in the containment sump

Question #137

Which of the following describes the methods for depressurizing the RCS in preparation for Refill in the order of preference used in 1BEP ES-1.2, “Post LOCA Cooldown and Depressurization”?

A. One Pzr PORV
Normal Spray
Aux Spray

B. Normal Spray
One Pzr PORV
Aux Spray

C. Normal Spray
Aux Spray
Two Pzr PORVS

D. Two Pzr PORVS
Normal Spray
Aux Spray

Question #138

Which of the following will satisfy conditions necessary to MANUALLY OPEN Containment Recirculation Valve SI8811A?

1. SI8812A - open
2. SI8812A - closed

3. CS001A - open
4. CS001A - closed

5. RH8701A - open
6. RH8701B - closed

A. 1, 3, 5

B. 2, 3, 5

C. 1, 4, 6

D. 2, 4, 6

Question #139

A plant heatup was in progress in accordance with BGP 100-1, when a leak was detected by the actuation of alarm “CNMT DRAIN LEAK DETECT FLOW HIGH.”

Following stabilization of the leak rate, the following plant conditions exist:

PZR level 42% and stable
PZR pressure 1600 psig and stable
Charging flow is 98 gpm as read on FI-121
Total letdown flow is 75 gpm
Total seal injection flow is 27 gpm
RCP seal parameters are normal

Which of the following actions will identify the correct leak location?

A. Closing the RCS loop drain valves will isolate a tube leak in the excess letdown heat exchanger.

B. Closing the orifice isolation valves and the letdown line isolation valves will isolate the leak downstream of 1CV131 letdown line pressure control valve.

C. Closing the individual seal injection isolation MOVs will isolate the leak at the seal injection line flange to the RCPs seal package.

D. Closing the charging line CNMT isolation valves will isolate the leak at the discharge line from the in service regenerative heat exchanger.

Question #140

Given the following plant conditions:

Plant in Mode 5
RCS temperature is 195°F and stable
RCS pressure is 325 psig and stable
Train “A” RH is in service, Train “B” RH is inoperable (OOS for repairs)
RCS is intact
All systems aligned in normal configuration for present conditions

A loss of RH shutdown cooling occurs with the temperature rising, which of the following is the preferred method for heat removal in accordance with 1BOA PRI-10?

A. RWST gravity feed to RCS, spill through the PZR PORVS

B. SI Pump Hot Leg Injection with spill through the 2-inch vent.

C. Natural or forced RCS flow while steaming intact S/Gs.

D. Reflux cooling to any S/G with level equal to or greater the 27% NR level.

Question #141

The following plant conditions exist on Unit 2:

A load reduction from 32% power was initiated 5 minutes ago
Current reactor power is 28%
PZR pressure 2235 psig and stable
PZR level 30% and stable
S/G levels (NR) 37%A, 39%B, 37%C, 38%D and stable

If the 2D S/G level were to drop to 29% and then rise to 35% 20 seconds later, what would be the response of the ATWS Mitigating System (AMS)?

A. AMS actuation signal is generated; the reactor trips and the motor driven AF pump start.

B. AMS actuation signal is generated; the main turbine would trip and both AF pumps start.

C. AMS actuation signal is NOT generated because turbine power is below C-20 setpoint.

D. AMS actuation signal is NOT generated because of a time delay in the S/G level circuit.

Question #142

Unit 1 is at 100% power with the following plant conditions:

Increased ouput on variable heaters
Letdown flow is 75 gpm
Charging flow is 105 gpm
S/G levels are constant
Tavg/Tref are matched

Which of the following events is in progress?

A. The PZR level control channel has failed high.

B. An atmospheric steam dump valve has opened.

C. A S/G tube leak has occurred.

D. PZR spray bypass flow has increased.

Question #143

Given the following:

Reactor power is 100%. .
RCS Tavg is stable at 582°F on all 4 loops.
RCS pressure is stable at 2235 psig.
Containment Pressure is INCREASING.
Steam Flow on each SG is STABLE
 1A SG Feed Flow is pegged HIGH
 1A SG Main FW Reg Valve is full OPEN
 1A SG pressure is STABLE
 1A SG level is DECREASING

Which of the following events is in progress?

A. Feed Line Break INSIDE Containment.

B. Steam Line Break INSIDE Containment.

C. Main FW Reg Valve failed OPEN.

D. Feed Flow Indicator pegged HIGH.

Question #144

The Unit is operating at 18% power. Which of the following describes the status of the Reactor Coolant Pump breakers and Reactor Trip breaker if the bus frequency for ALL RCP's is 55 Hz for 1 second?

	RCP Breakers	Reactor Trip Breakers
A.	Open	Shut
B.	Shut	Open
C.	Open	Open
D.	Shut	Shut

Question #145

Given the following Unit 1 conditions:

Reactor Power is 6%
Startup FW pump is in service
A and B CD/CB pumps are running
Instrument Air pressure is at 70 psig and dropping due to a header leak

Loss of air to which of the following COMPONENTS would result in an automatic reactor trip?

A. Condensate Pump recirculation valve CD152

B. CVCS Charging Flow Control valve CV121

C. RCP #1 Seal Leakoff Isolation valve CV8141A

D. Main FW Reg Bypass valve FW510A

Question #146

The following stable conditions are encountered when surveying a room located in the auxiliary building RPA:

General Area Radiation level in room	60 mrem/hr
Radiation level at 30 cm from pipe	375 mrem/hr
Radiation level on contact with pipe elbow	400 mrem/hr
Contamination levels	850 dpm/cm2 beta-gamma
	0 dpm/cm2 alpha
Airborne radiation level	0.6 DAC

What are the correct radiological postings or labels required to reflect the current radiological conditions for this room?

A. “DANGER, HIGH RADIATION AREA”
“HOT ZONE”
“CAUTION, CONTAMINATED AREA”.

B “CAUTION, RADIATION AREA”
“HOT ZONE”
“CAUTION, CONTAMINATED AREA”.

C. “DANGER, HIGH RADIATION AREA”
“HOT SPOT”
“AIRBORNE RADIOACTIVITY AREA”.

D. “CAUTION, RADIATION AREA”
“HOT SPOT”
“AIRBORNE RADIOACTIVITY AREA”.

Question #147

An operator received radiation exposure at both Braidwood and Byron Stations during the year.

The exposure record until the last day of the year is:

	Braidwood	Byron
Deep Dose Equivalent (DDE)	275 mrem	75 mrem
Lens Dose Equivalent (LDE)	15 mrem	10 mrem
Committed Effective Dose Equivalent (CEDE)	120 mrem	25 mrem
Shallow dose Equivalent (SDE)	25 mrem	15 mrem
Committed Dose Equivalent (CDE)	25 mrem	5 mrem

On the last day of the year the individual, at Byron Station, was requested to work in an area where the known radiation rate is 280 mR/hr. The source of the radiation is a nearby HOT SPOT inside a pipe trap where crud has been collecting and it has been determined to be totally gamma radiation.

If the worker takes 15 minutes to complete the task, what is the individual's Total Effective Dose Equivalent (TEDE) for the year?

A. 450 mrem

B. 565 mrem

C. 595 mrem

D. 660 mrem

Question #148

Unit 1 was operating at 28% power when the Loop B Reactor Coolant Pump (RCP) tripped on overcurrent.

Which of the following describes the unit's initial response? (Assume NO operator action AND NO rod motion.)

A. A reactor trip occurs and unaffected loop Tave increases.

B. A reactor trip occurs and unaffected loop Tave decreases.

C. A reactor trip will NOT occur and unaffected loop Tave decreases.

D. A reactor trip will NOT occur and unaffected loop Tave increases.

Question #149

Given the following plant conditions:

A LOCA has occurred on Unit 1
Power has been lost to BUS 142
The crew is initiating 1BEP ES-1.3 “Transfer to Cold Leg Recirculation Alignment”

Which of the following describes the affect of the loss of this bus on the Cold Leg Recirculation Alignment?

The SI pump’s suction is supplied by ......

A. both “A” train RH and “B” train RH from redundant paths.

B. “A” train RH only via crosstie to the CV Pumps suction valve 1CV8804A.

C. “B” train RH only via 1B RH discharge supply.

D. “A” train RH only via crosstie to the SI pumps suction valve 1SI8804B.

Question #150

Given the following plant conditions:

Unit 1 Reactor power was 85% with all control systems in automatic
1A MFP tripped .
The operator initiated a turbine runback

What was the initial response of the PZR pressure control system during this event?

A. The PORVs were blocked from opening to maintain pressure above the low reactor trip setpoint.

B. The variable heaters and the backup heaters turn full on to raise pressure to normal.

C. PZR Spray valves will throttle open to reduce pressure to normal.

D. Both PZR PORVs open to maintain pressure below the high reactor trip setpoint.

Question #151

Pressurizer Level is programmed from auctioneered-high __(1)__. Without program level, the pressurizer volume is _(2)_ to accommodate reactor coolant system water volume changes while limiting pressure transients.?

	(1)	(2)
A.	Tave	insufficient
B.	Tave	adequate
C.	Tc	adequate
D.	Tc	insufficient

Question #152

The following rod position indications exist:

The DATA B failure light is lit
LED for 24 steps is lit

What will be the range of the rod, using the normal and maximum indication accuracies due to coil placement and thermal expansion?

A. 20-32

B. 18-30

C. 16-30

D. 14-28

Question #153

A small break LOCA occurred coincident with loss of offsite power. The actions of 1BEP ES-1.2 Post LOCA Cooldown and Depressurization, are being performed. Plant status is as follows:

1A CV pump and 1B SI pump are running
Pressurizer level is stable at 58%
RCS pressure is stable at 900 psig
RVLIS indicates that a void exists in the reactor vessel head

With RCS subcooling stable at 3°F, the operator turns on a set of backup heaters to raise subcooling margin.

Which of the following describes the expected RVLIS and pressurizer level response?

	RVLIS	Pressurizer
A.	Decrease	Increase
B.	Increase	Decrease
C.	Decrease	Decrease
D.	Increase	Increase

Question #154

The following plant conditions exist:

LOCA is in progress
Containment Spray actuated due to high containment pressure
Containment Spray signal has been reset
Actions of ES-1.3 “Transfer to Cold Leg Recirculation” have been completed
Containment pressure is now 17 psig

Offsite power is then lost and the D/G output breakers have just closed onto the ESF buses

How are the Containment Spray Pumps restarted?

A. The pumps will auto start 15 seconds following closure of the D/G output breakers.

B. The pumps will auto start 40 seconds following closure of the D/G output breakers.

C. The operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 15 seconds following closure of the D/G output breakers.

D. The operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 40 seconds following closure of the D/G output breakers.

Question #155

The normal containment purge system is DESIGNED to:

A. aid in temperature, pressure and humidity control during LOCA conditions.

B. backup the hydrogen recombiners and maintain the concentration <4% by volume.

C. reduce radiation levels in containment during normal reactor operation.

D. provide continuous access to containment after a planned reactor shutdown.

Question #156

If all fuel racks in the Spent Fuel Pool are filled with irradiated fuel assemblies, what is the MINIMUM boron concentration required to maintain a safe reactivity condition of less than 0.95 Keff? (Assume new Holtec SFP storage racks)

A. 0 ppm

B. 150 ppm

C. 300 ppm

D. 500 ppm

Question #157

What is the mechanism that MINIMIZES the effect of shrink on indicated narrow range level for the Unit 2 D-5 S/G's when a reactor trip occurs from 100% power?

A. The circulatory velocity in the downcomer increases causing a pressure decrease.

B. Constant tempering flow reduces the preheat requirements for the incoming feedwater.

C. The level program maintains mass constant in the S/G.

D. The location of the lower level tap experiences a rise in static pressure that tends to offset the drop in the steaming rate.

Question #158

A loss of the condenser air removal system has occurred. Which of the following is the expected sequence of alarms?

a. Condenser vacuum low, turbine trip, C-9 Bypass Permissive lights

b. C-9 Bypass Permissive lights, condenser vacuum low, turbine trip

c. Condenser vacuum low, C-9 Bypass Permissive lights, turbine trip

d. Turbine trip, C-9 Bypass Permissive lights, condenser vacuum low

Question #159

Which of the following conditions are required to MANUALLY close the SAT feed on a 6.9KV breaker?

b. No lockouts on SAT or UAT feed

c. All SAT trips are in

d. UAT Feed Brkr C/S in A/C

e. UAT Feed Brkr open

Question #160

Which of the following identifies ALL the Fire Protection Pumps that will be running if system water pressure falls to 128 psig?

A. Diesel Engine Fire Pump, Electric Motor Driven Fire Pump, and both Jockey Pumps (0A and 0B).

B. Electric Motor Driven Fire Pump and the 0B Jockey Pump.

C. Diesel Engine Fire Pump and the 0A Jockey Pump.

D. Electric Motor Driven Fire Pump and both Jockey Pumps (0A and 0B).

Question #161

The following plant conditions exist on Unit 2:

The “0" CC HX is in service with the 2A CC Pump running
CC Surge Tank level was at 55% and is now at 60%
“0" CC HX Radiation Monitor RE-PR009 HIGH radiation level alarm is in

Which of the following describes the response of the CC system for these conditions?

A. No automatic actions occur.

B. The CC Surge Tank Vent Valve 2CC017 will automatically close and 1CC017 remains open.

C. The CC Surge Tank will be automatically isolated from letdown, prior to the CC Surge Tank completely filling and pressurizing.

D. Both CC Surge Tank Vent Valves, 1/2CC017, will automatically close.

Question #162

The following conditions exist for a job to be performed on a system.

The general area radiation levels are 10 mrem/hr in the room.
The hot spot in the room is a pipe elbow that has radiation levels of 100 mrem/hr.
The job will be performed near the hot spot area.

(Assumptions: ALL 4 cases below have the same transition time to and from destinations. All shielding placement and removal is at 100 mrem/hr)

Choose the method that best reduces personnel exposure.

A. Two Radiation Control personnel hang and remove 1 tenth thickness of lead shielding on the hot spot in 1.5 hours for the job. The job is performed after the lead shielding is in place by using 2 operators for 3 hrs each on the job.

B. The job is performed by 3 operators for 1 hr each on the job at the hot spot and a fourth operator reading instructions in the general room area for 1 hr.

C. The job is performed by 2 operators for 2 hrs each on the job at the hot spot and a third operator reading instructions in the general room area for 2 hrs.

D. The job is performed by using 2 operators for 3 hrs each on the job at the hot spot.

Question #163

Given the following conditions on Unit 1:

A LOCA has occurred.
The crew is in EP-0 at step 15 with the following plant conditions:

CETCs are reading 1090 °F
RCS pressure is 1950 psig
Containment pressure 6 psig and increasing
S/G pressures are 1180 psig
AFW maximum flow capability 400 gpm
S/G levels (NR): 1A S/G 25%, 1B S/G 24%, 1C S/G 26%, 1D S/G 30%

Based on the above conditions, what is the proper procedure to be in?

A. FR-C.1, "Response to Inadequate Core Cooling"

B. FR-H.1, "Response to a Loss of Secondary Heat Sink"

C. FR-Z.1, "Response to High Containment Pressure"

D. Transition to EP-1, "Loss of Reactor or Secondary Coolant"

Question #164

Unit 1 is at 100% power. Which of the following describes the plant response if the controlling pressurizer level channel fails HIGH with NO operator action taken?

A. The PZR heaters trip and letdown isolates on low level. The reactor eventually trips on actual high PZR level.

B. PZR level decreases until the reactor trips on low pressure. Letdown then isolates when level drops to 17%.

C. PZR level decreases initially, but stabilizes below the programmed setpoint. The controller will then restore level to program with an appropriate time constant.

D. The PZR heaters trip and letdown isolates on low level. The PZR will then gradually fill until a high pressure reactor trip occurs.

Question #165

The plant was operating at 10% Reactor Power when a loss of offsite power caused the RCPs to trip. Identify ALL of the indications that verify natural circulation is occurring.

1 - Core exit thermocouples --- decreasing
2 - Core exit thermocouples --- stable or increasing

3 - RCS hot leg temperature --- stable or decreasing
4 - RCS hot leg temperature --- increasing

5 - RCS subcooling --- decreasing
6 - RCS subcooling --- increasing

7 - RCS cold leg temperature --- at saturation for SG pressure
8 - RCS hot leg temperature --- at saturation for SG pressure

A. 1, 4, 5, 7

B. 2, 4, 6, 8

C. 1, 3, 6, 7

D. 2, 3, 5, 8

Question #166

Which of the following statements explains the BEP-1, "Loss of Reactor or Secondary Coolant," bases for stopping the RCPs as directed by the Operator Action Summary page following a containment Phase B actuation?

A. Delays the onset of two phase flow.

B. Preempt the RCP's tripping on cavitation because it is assumed that if containment spray actuates, an RCS depressurization is in progress.

C. Reduces the containment high pressure transient by lowering the energy release rate to containment from forced flow.

D. Precludes RCP bearings and seals from overheating on loss of component cooling water.

Question #167

Which of the following list ALL administrative requirements and interlocks associated with opening cold leg recirculation valves SI8811A and SI8811B.

A. No SI signal present
RWST level 45%
4 sump lights lit for RHR Pump NPSH.

B. SI signal present
RWST level 45%
2 sump lights lit for RHR Pump NPSH.

C. No SI signal present
RWST level 46%
2 sump lights lit for RHR Pump NPSH.

D. SI signal present
RWST level 46%
4 sump lights lit for RHR Pump NPSH.

Question #168

If the Reactor Coolant Subcooling Margin Monitor is not working properly, how will the subcooling margin be calculated?

A. Use 5 highest CETC average and RCS wide range pressure to determine subcooling margin.

B. Use 5 lowest CETC average and RCS wide range pressure to determine subcooling margin.

C. Use 10 highest CETC average and RCS wide range pressure to determine subcooling margin.

D. Use 10 lowest CETC average and RCS wide range pressure to determine subcooling margin.

Question #169

Which of the following is a positive indication that the PRT has ruptured following a pressurizer PORV failing full OPEN?

A. PRT temperature is decreasing.

B. PORV relief line temperature is increasing.

C. PRT level decreases to its normal value of 70%.

D. Pressurizer level is decreasing.

Question #170

Both units are at 100% power. The Component Cooling (CC) system is in its alignment for normal operations with ALL equipment operable.

A leak occurs resulting in the following conditions on Unit 2:

Alarm window for CC SURGE TANK LEVEL HIGH LOW actuates.
CC Surge Tank level is 33% and slowly falling
Demin Water and Primary Water makeup valves indicate OPEN
RCS temperature (average Tave) is 582°F and stable
PZR level is 60% and stable
VCT level is 42% and stable
Charging and letdown flows are balanced and normal Spent Fuel Pool level is stable

Where is the location of the CC System leak?

A. The seal water heat exchanger

B. The 2A RH pump seal cooler

C. The 2B letdown heat exchanger

D. The 2B excess letdown heat exchanger

Question #171

Which of the following is an indication that recombination is occurring after having placed the Hydrogen Recombiners in service?

A. Hydrogen Recombiner power increases to 20 KW.

B. Containment dewpoint decreases after Hydrogen Recombiners are placed in service.

C. Hydrogen Recombiner average thermocouple temperature is at or above 1200°F.

D. Containment pressure ~~deceases~~ *decreases* after Hydrogen Recombiners are placed in service.

Question #172

Which of the following is a function of the Service Air system?

A. to supply air to the Instrument Air system

B. to supply air to the primary emergency breathing air system

C. to supply air to only essential components

D. to supply oil filled compressed air for maintenance use

Question #173

During a cooldown on Unit 1 the following conditions exist:

RCS loop Tave: Loop 1: 550°F decreasing
Loop 2: 548°F decreasing
Loop 3: 551°F decreasing
Loop 4: 548°F decreasing
Steam header pressure- 1030 psig and decreasing
Steam Dump Mode Selector switch-STM PRESS MODE
Steam Dump Controller-MAN set at 30% demand

The operator momentarily places the Train A and Train B Steam Dump Bypass Interlock switches to Bypass and then releases them.

What is the status of the Steam Dump valves following the operator's actions?

A. All valves are fully closed

B. Three valves in group 1 are partially open

C. Three valves in group 1 are fully open and valves in group 2 are fully shut.

D. Three valves in group 1 are fully open and three valves in group 2 are partially open.

Question #174

The primary basis for depressurizing all intact steam generators to atmospheric pressure in FR-C.1, "RESPONSE TO INADEQUATE CORE COOLING," is to:

A. insure core exit thermocouple temperatures are reduced to less than 700 °F.

B. reduce S/G pressure to increase feedwater flow.

C. reduce RCS pressure for establishing low-head safety injection.

D. enhance natural circulation cooling of the reactor core.

Question #175

Given the following plant conditions on Unit 1:

Reactor Power is 100%
Power Range Nuclear Instrument channel N41 failed
Actions are complete in accordance with BOA INST-1

How is the Quadrant Power Tilt Ratio (QPTR) determined?

A. Incore detectors must be used.

B. The 3 operable power range NIS channels are used.

C. The 3 operable power range NIS channel are used in conjunction with flux map of the quadrant with the failed power range NIS.

D. 4 power range NIS channel values are used with the average values for the 2 adjacent power range NIS channels used for the failed channel.

Question #176

While transferring a fuel assembly from the spent fuel pool to containment, the gearbox for the Transfer Cart failed while the cart is in the Transfer Tube. How can the Transfer Cart be removed from the Transfer Tube?

A. Use a speedwrench on the gear box.

B. Use a crane to pull on the emergency pull-out cable.

C. Cut one drive chain and restart the drive motor.

D. Use the containment side drive motor to pull the cart.

Question #177

A caution statement in 1BFR-C.2, "Response to Degraded Core Cooling," states that an SI accumulator injection may cause a red path condition in INTEGRITY and that 1BFR-C.2 should be completed before transition to 1BFR-P.1, "Response to Imminent Pressurized Thermal Shock Condition." The caution statement applies during depressurization, prior to transitioning to 1BFR-P.1.

Which of the choices below correctly describes the reason for this caution statement?

A. Responding to the INTEGRITY Red path at this time could result in a CORE COOLING Red path.

B. The INTEGRITY Red path is a higher priority than the one being pursued in 1BFRC-2.

C. Responding to the INTEGRITY Red path at this time could result in an INVENTORY Red path.

D. The INTEGRITY Red path will be corrected by continuing the actions of 1BFRC-2.

Question #178

Given the following conditions for Unit 1:

A reactor trip and SI occurred at 0700
RH system problems resulted in a loss of recirculation capability
Current time is 1350
RCS subcooling is 10 degrees
1A and 1B CV pumps are running (Assume equal flow from each CV pump)
High head SI flow is 350 gpm
1A SI pump flow is 110 gpm
~~AB~~ *1A* SI pump flow is 80 gpm

Which of the following identified the ECCS pumps that should remain running following determination of minimum SI flow for decay heat removal? (References are provided)

A. Both SI pumps

B. Both Charging pumps

C. 1A Charging pump and 1A SI pump

D. 1B Charging pump and 1B SI pump

Question #179

Given the following plant conditions:

Unit 1 reactor tripped 30 minutes ago due to a partially stuck open S/G safety valve
2 RCCAs from Shutdown Group B stuck in the mid-out position.

A loss of offsite power occurred concurrently with the reactor trip.
1B D/G is OOS
1A D/G is operating as expected.

Present RCS temperature at 538°F
A continuous cooldown rate of 15°F/HR

Which of the following is the correct operator actions?

A. Emergency borate using the 1B charging pump from RWST and maximize charging flow due to the 2 RCCA's not fully inserting.

B. Emergency Borate using the Boric Acid transfer pump due to the cooldown and 2 stuck out RCCA's.

C. Determine the Shutdown Margin within the next 30 minutes to be within the COLR limits due to the cooldown and the 2 stuck out RCCA's..

D. Emergency borate using the 1A charging pump from RWST and maximize charging flow due to the cooldown and 2 stuck out RCCAs.

Question #180

Given the following Unit 1 conditions:

The unit is in HOT STANDBY
CST level is 66%
A loss of offsite power has occurred
Steam is being released through the SG PORVs

Which of the following states the MAXIMUM capability of the CST under the current conditions?

The CST can supply the AF pumps for....

A. an immediate cooldown at 100°F/hr to Mode 4.

B. 2 hours followed by a cooldown at 50°F/hr to Mode 4.

C. an immediate cooldown at 50°F/hr to Mode 5.

D. 2 hours followed by a cooldown at 100°F/hr to Mode 5.

Question #181

Per BCA-0.0, certain Engineered Safeguards equipment control switches are placed in the PULL-OUT position. Which of the following events is prevented by this switch alignment?

A. An uncontrolled depressurization of the RCS

B. An uncontrolled start of large loads on safeguards AC buses

C. An uncontrolled cooldown of the RCS and possible reactor restart

D. An uncontrolled use of water that may be needed for long term cooldown

Question #182

In accordance with 2BOA PRI-5, “Control Room Inaccessibility,” which of the following is performed for a fire in the control room prior to leaving?

A. Stop the 2A and 2B reactor coolant pumps

B. Verify Feedwater isolation

C. Open both PZR PORVs

D. Verify Main Steamline isolation

Question #183

The crew is performing 1BEP ES-1.2 “Post Loca Cooldown And Depressurization”. The only available power source for the ESF busses are the diesel generators. The diesels were started and have been continuously loaded to 6000 KW at 1050 amps for 1 hour. By design, how long could the diesel generators remain running under the present conditions?

A. The diesels must be secured immediately

B. 1 hour

C. 1999 hours

D. Indefinitely

Question #184

During a High Reactor Coolant Activity event, which of the following is the criteria used to determine if the standby mixed bed demineralizer should be placed in service?

A. Dose equivalent I-131 greater than 1 microcurie/gram.

B. Gross radioactivity greater than 100/Ebar.

C. Chloride levels greater than 1.0 ppm.

D. Decontamination factor less than 10.

Question #185

Given the following plant conditions:

Unit 1 is in Mode 6.
Core reload is occurring in the containment
Alarm 1-6-C3, “Refueling Cavity Level High/Low” is lit
Alarm 1-1-C1, “Spent Fuel Pit Level High/Low” is lit
SER printout shows the refueling cavity level is low

You are the Unit 1 Control Room Supervisor. Which of the following describes your immediate required actions?

A. Notify Radiation Protection to perform BRE-EXP.5, “Fuel Handling Emergency”

B. Check Reactor Cavity Leak Detection Loop, 1LI-RF-010

C. Initiate filling the cavity using BOP-RH-8, “Filling the Reactor Cavity for Refueling”

D. Notify Fuel Handling Foreman to move any fuel assembly in transit into the spent fuel pool or into the reactor

Question #186

A feedwater transient caused the average loop temperature for Loop C to decrease to 548°F.

Which of the following correctly completes the statement below?

Loop C average temperature must be restored ___________________.

A. immediately AND the reactor be in in Mode 3 within 1 hour.

B. within 5 minutes OR be in Mode 3 within the next 7 hours.

C. within 30 minutes OR be in Mode 2 with Keff <1.0.

D. within 2 hours OR be in Mode 2 with Keff <1.0 within 6 hours.

Question #187

Given the following timeline: (Assume appropriate TS actions have been completed or in progress)

5/21/2000	Sunday	0600	ACB 1414, ESF Crosstie Bkr out of service
5/24/2000	Wednesday	0600	Electrical system demand is red. Post maintenance testing in-progress for ACB 1414. Unit remains online at 100% power
5/24/2000	Wednesday	0800	Plant operator discovers large pool of oil on floor in 1B DG room. 1B DG declared inoperable.
5/24/2000	Wednesday	1000	ACB 1414 is returned to service
5/27/2000	Saturday	0200	Major thunderstorm in area, knocks out Unit 2 SATs

If the 1B DG and/or Unit 2 SAT cannot be repaired within the appropriate LCO time period, when is the LATEST time the unit would be required per TS to be in MODE 3?

A. at 0800 on Saturday, 5/27/2000

B. at 1200 on Saturday, 5/27/2000

C. at 1400 on Saturday, 5/27/2000

D. at 2000 on Saturday, 5/27/2000

Question #188

Given the following conditions:

Unit 1 startup is in progress
Reactor Power is at 3% thermal power
IR channel ~~N-31~~ *N-35* failed low

To meet Technical Specification requirements, you:

A. MUST immediately suspend startup and lower power to below P-6.

B. MUST immediately trip the reactor.

C. MAY continue with startup but must ensure reactor power is above P-10 within 2 hours.

D. MAY continue with startup but cannot make the mode change until the ~~N-31~~ *N-35* is repaired.

Question #189

An accident is in progress on Unit 1. The following plant conditions exist:

Containment pressure 3 psig (slowly decreasing)
SG levels (Narrow range) A: 5% B: 7% C: 8% D:5% (All are slowly increasing)
Main steamline 1B radiation alert alarm is lit

In accordance with 1BEP-3, "Steam Generator Tube Rupture," you direct the operators to:

A. immediately manually CLOSE the 1B AF isolation valves, 1AF013 B and F.

B. maintain feed to the 1B S/G until narrow range is 10%, then manually isolate AF.

C. maintain feed until ALL S/G narrow range level are 10%, then manually isolate AF to the 1B S/G.

D. maintain feed to the 1B S/G until narrow range is 31%, then manually isolate AF.

Question #190

Given the following plant conditions:

Unit 1 is at 10% power
Preps are in progress to put the turbine online
1A and 1B Circ water pumps are operating
1C circ water pump is OOS for shaft repair

RCS temperature is 559 °F
Steam Dumps are in Steam Pressure Mode

Which of the following describes the effect on the Steam Dump system if a fault occurs on bus 143?

Steam dumps are armed....

A. but not controlling pressure.

B. and in the plant trip controller mode.

C. and remain in the steam pressure controller mode.

D. and in the load reject controller mode.

Question #191

While performing 1BEP ES-1.2 a void was drawn in the reactor vessel. The crew then entered 1BFR-I.3 "RESPONSE TO VOIDS IN REACTOR VESSEL" based on a yellow condition in the critical safety function status trees. The crew is venting the reactor vessel to eliminate the voids. Which of the following is **NOT** part of the venting termination criteria?

A. RVLIS Plenum indicates 85%.

B. PZR level < 21%.

C. RCS pressure decreases by 200 psi from starting pressure.

D. Venting time is > maximum calculated time.

Question #192

The detector for 1RT-AR011, Containment Fuel Handling Incident Train A Rad monitor, fails causing the output to go high.

Which of the following would occur due to this failure?

A. Upward movement of Refueling Machine hoist is inhibited.

B. Starts 0VA04CA, Fuel Handling Building Charcoal Booster Fan.

C. Closes 1VQ005A, Containment Mini Flow Purge Exhaust Isolation

D. Prevents 1PR039, Containment Atmosphere to Process Outside Isolation Valve from being opened.

Question #193

1BFR-Z.1, “Response to High Containment Pressure” contains a CAUTION which states to operate containment spray in accordance with 1BCA-1.1, “Loss of Emergency Coolant Recirculation,”(if applicable). 1BCA-1.1 determines the number of operating CS pumps based on which of the following?

A. Containment pressure, containment temperature, and sump level.

B. Containment pressure, operating RCFCs, and RWST level

C. Containment temperature, operating RCFCs, and RWST level

D. Containment pressure, operating RCFCs, and sump levels

Question #194

The limits on RCS activity provided in Technical Specifications are based on the dose that would be received at the site boundary in a SGTR accident that begins with steady-state primary-to-secondary leakage of 1 gpm. Maintaining these RCS activity limits ensures that the 2-hour dose at the site boundary during a SGTR will NOT exceed:

A. 10 CFR 20 “Standards for Protection Against Radiation,” limits

B. 10 CFR 100, “Reactor Site Criteria,” limits

C. EPA Protective Action Guideline thresholds

D. 5 Rem TEDE for the general public

Question #195

Unit 2 was operating at 100% power when a large break LOCA occurred. All safeguards equipment responded as designed. The crew has transitioned to 2BEP-1, “Loss of Reactor or Secondary Coolant.”

Which of the following radiation monitor alarms will be used by the Station Director in assessing the emergency action levels?

A. 2RT-AR001 (Containment Area)

B. 2RT-AR011 (Containment Fuel Handling Incident)

C. 2RT-PR011 (Containment Atmosphere)

D. 2RT-AR020 (Containment High Range)

Question #196

Unit 1 was at 100% power when the following events occurred:

- ALL SGs are faulted into the containment
- Upon transition from 1BEP-0 to 1 BEP-2, a RED path is noted on the containment critical safety function, so the actions of BFR-Z.1, “Response to High Containment Pressure” are performed.
- Auxiliary Feedwater has been throttled to 25 gpm to each steam generator
- When directed by BFR-Z.1 to return to procedure and step in effect, the following status is noted on the CSF status tress:

Subcriticality:	Green
Core Cooling:	Green
Heat Sink:	Red
Integrity:	Orange
Containment:	Red
Inventory:	Yellow

Which of the following ACTIONS will be performed next?

A. Isolate the faulted SGs in accordance with 1BEP-2, “Faulted Steam Generator Isolation”

B. Feed and Bleed in accordance with 1BFR-H.1, “Response to Loss of Secondary Heat Sink”

C. Perform RCS soak in accordance with 1BFR-P.1, “Response to Imminent Pressurized Thermal Shock Condition”

D. Verify containment spray lineup in accordance with 1BFR-Z.1, ”Response to High Containment Pressure”

Question #197

Steps 3 and 4 of BFR-S.1, "Response to Nuclear Power Generation/ATWS", require the operator to check AF pumps running and initiate Emergency Boration of RCS.

Which of the following is the reason that each of the above actions must be performed manually by the operator instead of through manual initiation of SI?

A. Initiation of SI will compound the problem by charging the RCS system solid, causing pressurizer PORVs and safety valves to lift.

B. Initiation of SI will create a loss of heat sink problem.

C. Initiation of SI will result in a turbine trip which is required for a heat sink.

D. Automatic initiation of SI is preferred but operator action is necessary to anticipate and mitigate the ATWS.

Question #198

Given the following plant conditions on Unit 1:

Reactor Trip and Safety Injection have occurred
MSIVs have just closed due to containment pressure
RCS pressure is 1700 psig and stable
CETCs indicate 570 °F
Total AFW flow is 700 gpm
PZR level is 42%

Based upon these conditions, the operators should _____ (Select ONE of the following)

A. verify all RCPs are stopped.

B. terminate Safety Injection.

C. transition to ES-0.0, “Rediagnosis.”

D. initiate containment spray as a result of increasing containment pressure.

Question #199

While reviewing the results of an SI pump discharge flow surveillance, you recall that the impeller had been modified during the last outage. You note that the procedure had not been revised to reflect the new flow rates which were discussed during the last requalification cycle training class. You notify the system engineer who confirms that the acceptance criteria should have been changed when the modification was closed out.

Which of the following actions are required?

A. Line out/initial the procedure steps and replace values with the new flow rates. Approve the surveillance results based on new values.

B. Approve the surveillance results based on the current procedure.

C. Obtain a second SRO review and approval of the procedure change prior to approving the surveillance results.

D. Obtain a permanent procedure change prior to approving the surveillance results.